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Revised Safety Analysis for the HEU to LEU Conversion of the Washington State University Reactor

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Safety Analysis for the HEU to LEU Conversion of the Washington State University Reactor

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Final Report

Complete Final HEU/LEU Conversion Report

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GENERAL ATOMICS
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QTRIGA Reactors

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Summary

This report contains the results of the design, safety, and accident analyses performed for Washington State University (WSU) by General Atomics for the conversion of the WSU 1 MW TRIGA research reactor. Currently the WSU Mixed HEU Core 34A contains a mixture of TRIGA fuel elements which include highly enriched (70% U-235) fuel elements developed as part of the Fuel Life Improvement Program (FLIP) and low enriched (<20% U-235) standard fuel elements. The conversion will replace the FLIP/HEU fuel elements with specially designed TRIGA LEU fuel elements containing 30 weight percent uranium enriched to less than 20 % in U-235 (30/20 LEU). The 30/20 LEU fuel elements are specially designed for a one-to-one replacement of the FLIP HEU fuel elements. By replacing the FLIP HEU fuel elements, all HEU will have been removed from the core and eventually from the facility. This study investigates the performance and safety margins of the proposed converted LEU core under nominal and accident conditions. It also identifies any suggested changes to the WSU Final Safety Analysis Report and Technical Specifications, Ref. 1.

1.0 General Description of the Facility

1.1 Introduction

This section provides an overview of the changes to the physical, nuclear and operational characteristics of the facility required by the HEU to LEU conversion of the WSU 1 MW TRIGA reactor.

The HEU to LEU conversion requires a one-to-one swap for each of the TRIGA FLIP HEU fuel elements currently operating in WSU Core 34A by specially designed TRIGA LEU (30 /20) fuel elements. WSU Core 34A is also operating with low enriched standard TRIGA Standard Fuel Element (SFE) consisting of 8.5 weight percent uranium / 20% enriched fuel (8.5/20) LEU. These latter fuel elements will not be replaced and will continue operating even after the replacement of the FLIP HEU fuel elements. In addition, the conversion will not require any changes to the remainder of the facility.

The proposed converted WSU Mixed LEU reactor core designated as Core 35A contains:

1. 49 – Fresh TRIGA (30/20 LEU) Fuel Elements whose fuel contains 30 weight percent uranium / < 20% enriched in U-235. These elements will be referred as 30/20 LEU SFEs
2. 2 – Fresh TRIGA (30/20 LEU) Instrumented Fuel Elements (IFEs) whose fuel contains 30 weight percent uranium / 20% enriched in U-235. These elements will be referred as 30/20 LEU IFEs
3. 68 – currently operating and partially burned TRIGA SFEs whose fuel contains 8.5 weight percent uranium / < 20% enriched in U-235. These elements will be referred as 8.5/20 partially burned LEU SFE's
4. 1 – new Transient Pulsing Rod, this will be referred as Rod 3
5. 3 – currently operating control rod blades. These blades will be referred to as Shim blade 1, Shim blade 2, Shim blade 4

6. 1 – currently operating stainless steel regulating blade. This will be referred to as Servo blade 5.

Based on this core configuration, it is concluded that:

i) the shutdown margin meets the required limits;
ii) the reactivity coefficients remain essentially the same as for WSU Mixed HEU Core 34A;
iii) fuel integrity of the converted core is maintained under all operating conditions; and
iv) dose to public from the maximum hypothetical accident (MHA) and fuel handling accident (DBA) remain essentially unchanged from the HEU core and below the maximum permissible limits.

The HEU to LEU conversion may require possible changes to the Technical Specifications as discussed in Section 14.

1.2 Summary and Conclusions of Principal Safety Considerations

The WSU Mixed LEU Core 35A meets all the safety requirements as specified in the 1979 Safety Analysis Report which is the current SAR in force. (Reference 1)

1.3 Summary of Reactor Facility Changes

The replacement 30/20 LEU 4-rod and 3-rod fuel clusters have the same physical dimensions as the currently operating FLIP HEU 4-rod and 3-rod clusters. The clad for the 30/20 LEU fuel and the construction of the clusters are identical with those currently used with the FLIP HEU fuel. The 30/20 LEU fuel has been approved by the Nuclear Regulatory Commission (NRC) for use in non-power reactors. (Reference 2)

1.4 Summary of Operating License, Technical Specifications, and Procedural Changes

In addition to the updated LEU fuel parameters, the maximum pulsed reactivity in the Technical Specifications may change (see Section 14).

1.5 Comparison with Similar Facilities Already Converted

The Texas A&M (TAMU) 1 MW TRIGA reactor has already been converted to TRIGA LEU (30/20) fuel. There are both similarities and differences between the WSU and TAMU reactors.

The similarities are primarily in the TRIGA SFEs and Clusters which are similar in configuration and design for both TAMU and WSU.

The differences are primarily in the core configuration. This includes:

- WSU will be a partial fuel replacement.
    - Fuel – For WSU, only the FLIP/HEU SFEs and IFEs will be replaced whereas TAMU was a full FLIP/HEU core conversion including the replacement of all SFEs, and IFEs.
    - Control – For WSU only the Transient Rod (TR) (Rod 3) will be replaced whereas for TAMU the entire control system consisting of the Fuel Followed Control Rods (FFCR), TR, and Regulating Rod (RR) was replaced.

- The WSUs control system contains 3 control (shim) blades, a stainless steel regulating blade (servo blade) and a TR (water followed) whereas the TAMU control system has 4 standard TRIGA FFCRs, an RR and a TR (air followed).
- WSU has a uniform pitch between Fuel Clusters whereas the TAMU fuel cluster pitch is different in the two directions. Table 1 summarizes these similarities and differences between the two reactors.

Table 1 Comparison of WSU and TAMU Conversions

Component	TAMU	WSU
Conversion Core	Full Conversion: 30/20 LEU	Partial Conversion: 30/20 LEU & Partially Burned 8.5/20 LEU
Standard Fuel Element	Same	Same
Instrumented Fuel Element	Same	Same
Fuel Rod Clusters	Same	Same
Fuel Cluster Pitch	Variable in two directions	Uniform in the horizontal plane
Control Rods	4 TRIGA FFCRs	3 Control (Shim) blades
Regulating Rod	TRIGA RR – water followed	Stainless steel (Servo) blade
Transient Rod	TRIGA TR – air followed	TRIGA TR – water followed

2.0 Site Characteristics

The HEU to LEU conversion does not impact the site characteristics.

3.0 Design of Structures, Systems, and Components

The HEU to LEU conversion does not require any changes to the design of structure, systems, and components.

4.0 Reactor Description

4.1 Reactor Facility

The HEU to LEU conversion of the WSU facility requires only changes in the fuel type. All the following aspects of the facility remain unchanged:

- Neutron Reflector
- Neutron Source and Holder
- Reactor Tank and Biological Shielding
- Core Support Structure
- Functional Design of the Reactivity Control System

Table 2 provides a comparison of the key design safety features of the HEU and LEU fuel clusters and a comparison of the key reactor and safety parameters that were calculated for each core. The results show that the WSU reactor facility can be operated as safely with the addition of the new LEU fuel clusters as with the present HEU fuel clusters.

The evaluation of WSU Mixed HEU Core No 34A provides an opportunity to benchmark the computational technique to be used for evaluating the converted WSU Mixed LEU Core 35A. The analyses produced operational parameters to be compared with the actual measured values from the operational data conducted by WSU Staff for Core 34A. The experimentally measured parameters included the reactivity for the fully loaded Core 34A which contains 51 FLIP HEU SFEs/IFEs and 68 partially-burned 8.5/20 LEU SFEs with full water reflection; the control rod calibration values; the reactivity loss and peak fuel temperatures as a function of reactor power; and pulsing performance including peak power, peak fuel temperature, and energy production all as a function of prompt reactivity insertion. In addition, the computation was used to produce a plot of the prompt, negative temperature coefficient of reactivity ($\Delta k/k$-°C) versus reactor fuel temperature that can be compared with the value in the SAR (1979) for the same parameter.

The steady state parameters for the WSU Mixed LEU Core 35A were calculated using the same computational procedures adapted to the WSU Mixed HEU Core 34A configuration.

Table 2 Washington State University HEU – LEU Conversion Design Data

	Mixed HEU Core 34A		Mixed LEU Core 35 A	
FUEL PARAMETERS				
	FLIP HEU SFEs/IFEs	8.5/20 LEU SFEs	30/20 LEU SFEs/IFEs	8.5/20 LEU SFEs
Number of Fuel Rods	51	68	51	68
Fuel Type	$UZrH_x$	$UZrH_x$	$UZrH_x$	$UZrH_x$
Uranium Weight/Enrichment, %	8.5/70	8.5/20	30/20	8.5/20
Erbium, weight %	1.48	--	0.90	--
Zirconium Rod, OD, mm	6.35	6.35	6.35	6.35
Fuel meat, OD, mm	34.823	34.823	34.823	34.823
Fuel meat length, mm	381	381	381	381
Clad Material	304 SS	304 SS	304 SS	304 SS
Clad thickness, mm	0.508	0.508	0.508	0.508
REACTOR PARAMETERS				
Reactor Power				
Licensed Power, MW	1.0		1.0	
LCO Max Power, MW	1.3		1.3	
Max Fuel Temperature at 1 MW, °C	435		500	
Calculated Maximum Pulsing Operation with $\hat{T}$ limited to 830 °C, MW (a)	\$2.02		\$2.04 BOL \$2.20 EOL	
Cold Clean Excess Reactivity, Δk/kβ, \$	7.17		6.94	
Prompt Negative Temp. Coefficient of Reactivity -Δk/k-°C 23-1000°C	0.54×10^{-4} to 1.51×10^{-4}		0.60×10^{-4} to 1.27×10^{-4}	
Coolant Void Coefficient, Δk/k per 1% void	- 0.080% (b)		- 0.135% (b)	
Maximum Rod Power at 1 MW, kW/element	20.9		20.8	
Average Rod Power at 1 MW, kW/element	8.4		8.4	
Maximum Rod Power at 1.3 MW, kW/element	27.2		27.0	
Average Rod Power at 1.3 MW, kW/element	10.9		10.9	
Maximum Rod Power at DNB = 1.0, kW/element	51.7		52	
DNB Ratio at Operating Power	2.47		2.45	
Prompt Neutron Lifetime, μsec	30.7		28.2	
Effective Delayed Neutron Fraction	0.0076		0.0075	
Shutdown Margin, Δk/kβ (\$)with most reactive rod and Reg. Rod Stuck out	\$2.61		\$2.82	
SAFETY PARAMETERS				
Limiting Safety System Setting	500°C		500°C	
LCO Max Power, MW	1.3		1.3	
Minimum DNB ratio at 1.0 MW	2.47		2.45	
Minimum DNB ratio at 1.3 MW	1.90		1.89	
Calculated Maximum Positive Pulsed Reactivity Insertion to reach $\hat{T}$=830°C, Δk/kβ (\$) (a)	\$2.02		\$2.04 BOL \$2.20 EOL	
Peak Pulsed Fuel Temperature, °C	830		830	

(a) Calculated maximum reactivity insertion for pulsing yields a conservative value, as shown by actual pulsing data.
(b) Coolant void coefficient of reactivity is a negative reactivity.

4.2 Reactor Core

This chapter provides a description of the components and structures in the reactor core. Comparisons between the mixed HEU and mixed LEU cores are presented when the conversion requires changes in some characteristics.

The WSU reactor is primarily a homogeneous, light water moderated and cooled, tank-type reactor fueled with a core containing a mixture of FLIP HEU SFEs and 8.5/20 LEU SFEs in either a 4-rod or 3-rod cluster configuration. The fuel clusters are supported by a grid box consisting of a cast aluminum grid plate suspended from the bridge by four corner posts that form a suspension frame. The grid plate provides a 7 × 9 array of square holes for fuel clusters and two slots for control blades. The grid box accepts the 4-rod and 3-rod fuel clusters and the reflector elements. The reactor is supported from the top of the biological shield and is moveable along the central, long axis of the reactor tank. Figure 1 shows the WSU movable reactor core. At WSU, the most frequently used reactor locations are: (1) adjacent to the thermal column (BNCT filter) as shown in Figure 1, and (2) about 7 feet removed from the BNCT filter and water reflected. The arrangement for WSU Core 34A, a Mixed HEU core configuration is shown in Figure 2. It contains a mixture of 51 FLIP HEU SFEs and 68 – 8.5/20 LEU SFEs, 3 control blades, a servo blade, and a water-followed transient rod. All 5 control rods are supported from the bridge structure at the top of the biological shield.

Figure 1 Pool Structure at the Nuclear Radiation Center – Washington State University

Figure 2 WSU Mixed HEU Core 34A

4.2.1 Fuel Elements

The FLIP HEU, the partially burned 8.5/20 LEU, and the 30/20 LEU fuel elements have similar overall designs, i.e. they are all TRIGA fuel rods mounted in 4-rod or 3-rod, clusters.

Fuel Description

The geometries, materials, and fissile loadings of the current FLIP HEU SFEs, the partially-burned 8.5/20 LEU SFEs, and the 30/20 LEU SFEs are summarized in Table 3.

Figure 3 shows a 4-rod fuel cluster and Figure 4 shows a 3-rod cluster modified to accommodate the water followed transient rod. Figure 5 shows the nominal cluster and fuel element spacing.

The heat removal system for the WSU Mixed LEU Core 35A remains unchanged from that used with the Mixed FLIP HEU Core 34A. The primary cooling system circulates heated water from the reactor tank through the heat exchanger and returns the cooled water to the reactor tank. The secondary cooling system circulates water from the heat exchanger to the cooling tower. The core itself is cooled by natural convection.

The 30/20 LEU SFEs to be installed in the WSU core are contained within 4-rod and 3-rod fuel clusters exactly the same as for the present Mixed HEU core. Figures 6 and 7 show detailed illustrations of the fuel element and the instrumented (integrated thermocouple) fuel element. The fuel for the FLIP HEU, the partially burned 8.5/20, and the 30/20 LEU SFEs differs only in the alloy in the fuel sections of these illustrations; the dimensions are the same for these types of TRIGA fuel.

Table 3 Description of TRIGA HEU and LEU Fuel Elements

Design Data	TRIGA FLIP HEU	TRIGA 30/20 LEU	TRIGA 8.5/20 LEU
Number of Fuel Elements at Full Load	51	51	68
Fuel Type	U-ZrH (FLIP)	U-ZrH (30/20)	U-ZrH (8.5/20)
Enrichment, %	70	19.75	19.75
Uranium Density			
g/cm^3		2.14	0.5
wt-%	8.42	30	8.5
Number of Fuel Elements per Cluster	4	4	4
^{235}U per Fuel Bundle, g		597.26	156
^{235}U per Fuel Element, g		149.32	39
^{166}Er per Fuel Element, g	10.27	7.46	----------
^{167}Er per Fuel Element, g	7.09	5.15	----------
Erbium, wt-%	1.48	0.9	----------
Zirconium Rod Outer Diameter, mm	6.35	6.35	6.35
Fuel Meat Outer Diameter, mm	34.823	34.823	34.823
Fuel Meat Length, mm	381	381	381
Cladding Thickness, mm	0.508	0.508	0.508
Cladding Material	304 SS	304 SS	304 SS

Figure 3 Four Element Cluster

Figure 4 Three-Element Cluster with Guide Tube for Transient Rod

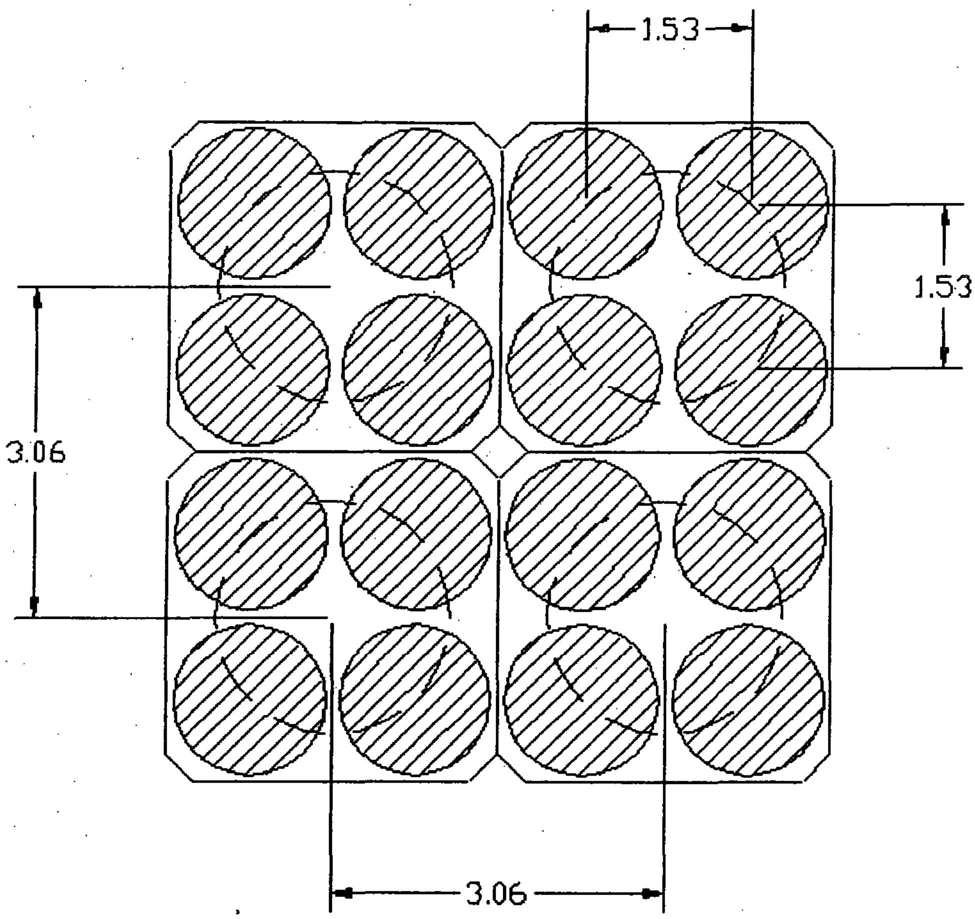

Figure 5 Nominal Fuel Element and Cluster Spacing in the WSU Core (dimensions are inches)

Figure 6 Detailed Drawing of Fuel Element

Figure 7 Instrumented Fuel Element

4.2.2 Control System

The three shims and the servo control elements of the WSU reactor are blade type elements as shown in Figure 8. The poison section of the safety blades is a boral sheet (35 wt.% B_4C, 65 wt.% Al) 40.5 inches long and 10.5 inches wide. The boral sheet is 3/8 inches thick and is clad with 1/8 inch aluminum. The regulating servo blade is a stainless steel sheet about 11 inches wide and 40 inches long. Each blade is guided through its travel by a shroud, as shown in Figure 8. The shroud consists of two thin aluminum plates 38 inches high separated by aluminum spacers to provide a 3/4 inch control blade slot. Small flow holes are drilled at the bottom of the shroud to reduce the effects of viscous damping on the blade fall time. There is no plan to change either the WSU shim blades or the servo control blade.

The transient rod (TR3) is a standard TRIGA water-followed transient rod and will be replaced during the conversion, Figure 9.

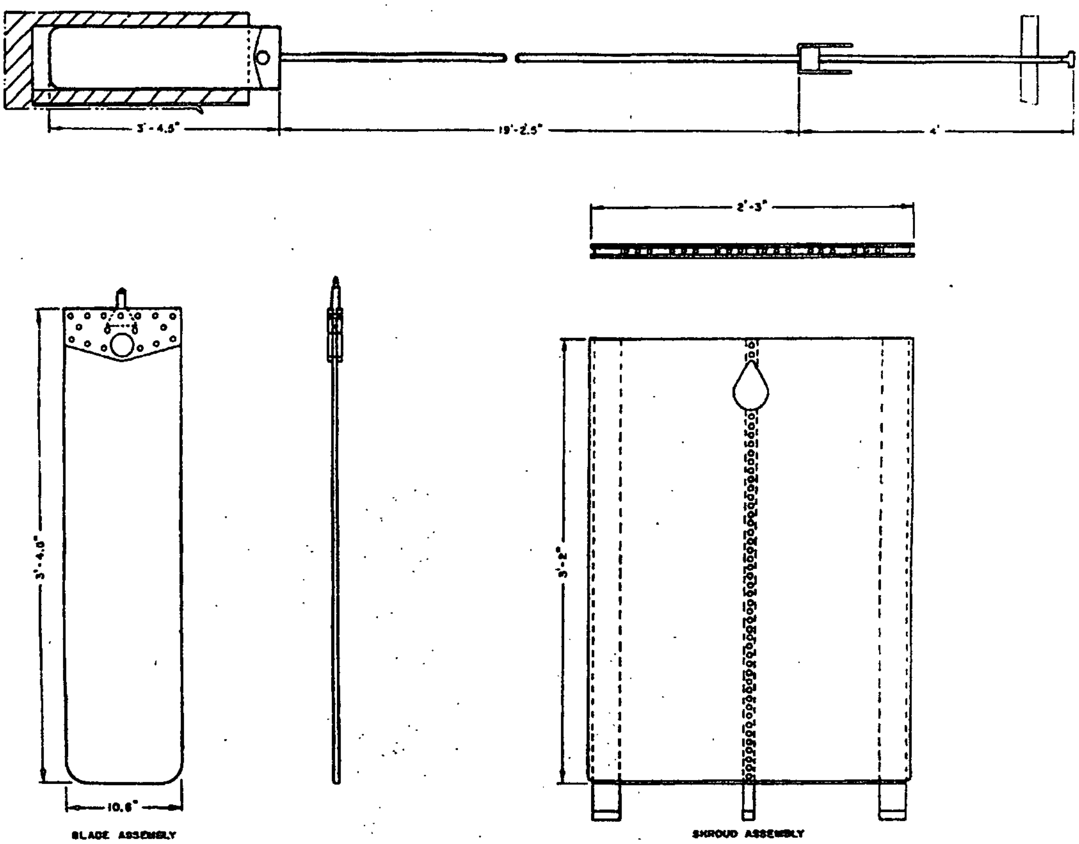

Figure 8 Control Blade

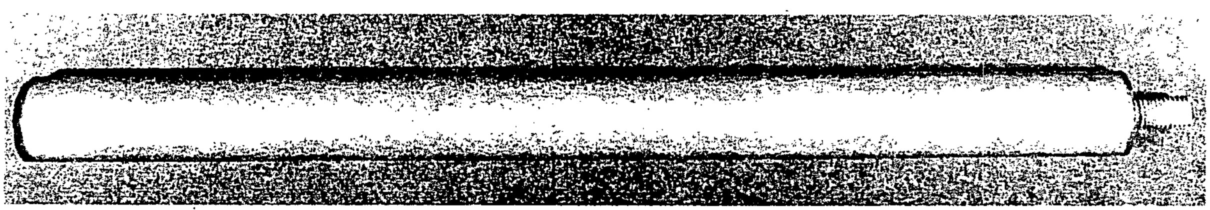

Figure 9 Standard TRIGA Water Followed Transient Rod

4.2.3 Neutron Reflector

There is no plan to change to a different type of reflector during the HEU to LEU conversion. The WSU reactor uses nuclear-grade graphite elements (as well as water) in the core as reflector, Figures 1 and 2 show the reflector regions.

4.2.4 Neutron Source and Holder

The proposed HEU to LEU conversion of the WSU core does not require any changes in the existing neutron source location, Fig. 4.2.

4.2.5 In-Core Experimental Facilities

There are no in-core experiments in the WSU reactor.

4.2.6 Reactor Materials

The WSU conversion to LEU requires a change in the fuel element composition but no change in the fuel clad. Table 4 presents the material composition used in the computational models.

Table 4 Material Composition used in the MCNP Models

Material	Nuclide	Nuc. Den. (atoms/b-cm)	Physical Density (g/cc)
SS 304 (clad)	Cr-50	0.000778	7.98
	Cr-52	0.015003	
	Cr-53	0.001701	
	Fe-56	0.056730	
	Ni-58	0.007939	
	Mn-55	0.001697	
Graphite (reflector in fuel)	C		1.75
Zirconium (rod)	Zr		6.51
6061 Al (grid plate and control rod clad)	Al-27	0.058693	
	Fe-56	0.000502	
90% B_4C (transient rod)	B-10	0.020950	
	B-11	0.084310	
	C	0.026320	
Boral (35wt% B_4C 65 wt% Al)	B-10	0.008058	
	B-11	0.032233	
	C	0.010073	
	Al-27	0.038306	
Al+Water Mix 1 (2" lower cluster adapter)	H	0.028748	
	O	0.014374	
	AL-27	0.033455	
	FE-56	0.000286	
Al+Water Mix 2 (5" grid plate)	H	0.030788	
	O	0.015394	
	AL-27	0.031663	
	FE-56	0.000271	
Water			1.0
Air			0.000123

4.3 Reactor Tank and Biological Shielding

The proposed HEU to LEU conversion of the WSU core does not require any changes in the reactor tank or biological shielding.

4.4 Core Support Structure

The proposed HEU to LEU conversion of the WSU core does not require any changes in the core support structure.

4.5 Dynamic Design

4.5.1 Calculation Models: Nuclear Analysis Codes

Three-dimensional calculations are performed using both diffusion theory and Monte Carlo codes. In general, multi-group diffusion theory is used for design calculations since it gives adequate results for systems of this kind and its multi-group fluxes and cross sections are easily utilized in nuclide burnup calculations. The Monte Carlo calculations are used to evaluate the facilities around the core and also to compute the worth of core components and different core configurations.

The diffusion theory code is DIF3D, a multi-group code which solves the neutron diffusion equations with arbitrary group scattering. (References 3 and 4)

The Monte Carlo code is MCNP5 that contains its own cross section library. (Reference 5)

The BURP/DIF-3D module, (Reference 6), is used for the burnup calculations with the cross section data generated with GGC-5, (Reference 7).

MCNP5 Monte Carlo Code

This section discusses the MCNP5 models developed for these analyses and the benchmark calculations for the Mixed HEU Core 34A, and determines a reference critical Mixed LEU Core 35A.

Reactor calculations were performed in three dimensions for the full core loading of the WSU Mixed HEU Core 34A and for the initial criticality and the full core loading of the WSU Mixed LEU Core 35A using the MCNP 5, Version 1.3, continuous energy Monte Carlo code. The nuclide cross sections were based on ENDF/B VI data included in the MCNP 5 data libraries.

To obtain the heavy metals and fission products densities in the partially-burned 8.5/20 LEU and FLIP HEU fuel rods in Core 34A the three dimensional depletion calculations were performed with BURP-DIF3D codes for each of FLIP and STD fuels. The FLIP fuels were burned 3000 days at 1 MW, and the STD fuels were burned 1500 days at 1 MW. The mass of U-235 and U-238 in Core 34A were used from the September 2005 WSU inventory report listed in Table 5.

The heavy metal masses and fission products were calculated for the WSU Mixed HEU Core 34A. To make the graphs easier to read the results are plotted as a function of burnup days along with polynomial fits for the nuclides in Figures 10 through 15

By using the U-235 mass in the inventory report and the U-235 polynomial fit, the burnup days for each fuel rod were calculated. The obtained burnup days were used in the polynomial fits to calculate the heavy metals and fission products in the depleted fuel rods.

The nuclide densities used in the MCNP models are shown in Table 6 for the WSU FLIP HEU fuel meats, in Table 7 for the burned 8.5/20 LEU fuel meats, and Table 8 for the 30/20 LEU fuel meats.

The other materials excluding the fuel used in the Mixed HEU and Mixed LEU MCNP models are listed in Table 4.

Table 5 Uranium inventories for the partially-burned 8.5/20 LEU (STD) and FLIP HEU fuel rods – WSU Mixed HEU Core 34A, as of September 30, 2005

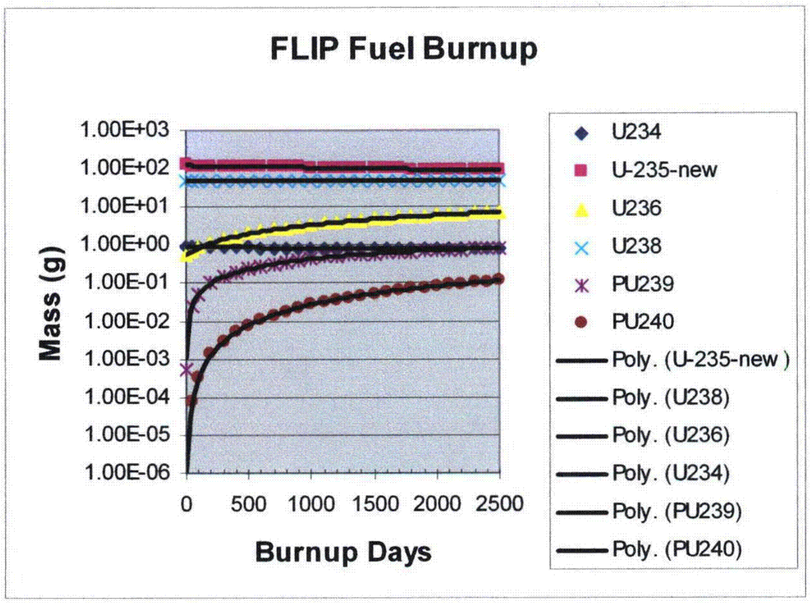

Figure 10 Masses for Typical Heavy Metals and Fission Products – Core 34A

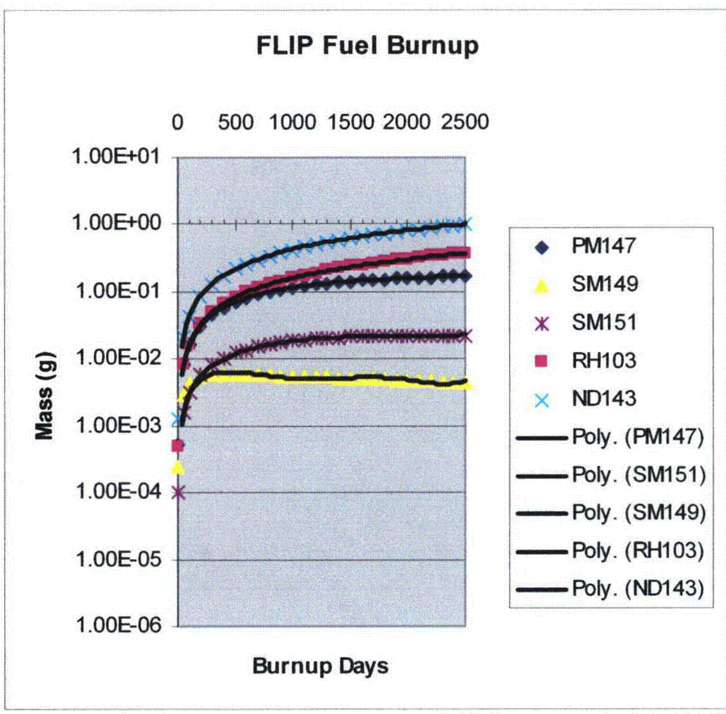

Figure 11 Masses for Typical Heavy Metals and Fission Products – Core 34A

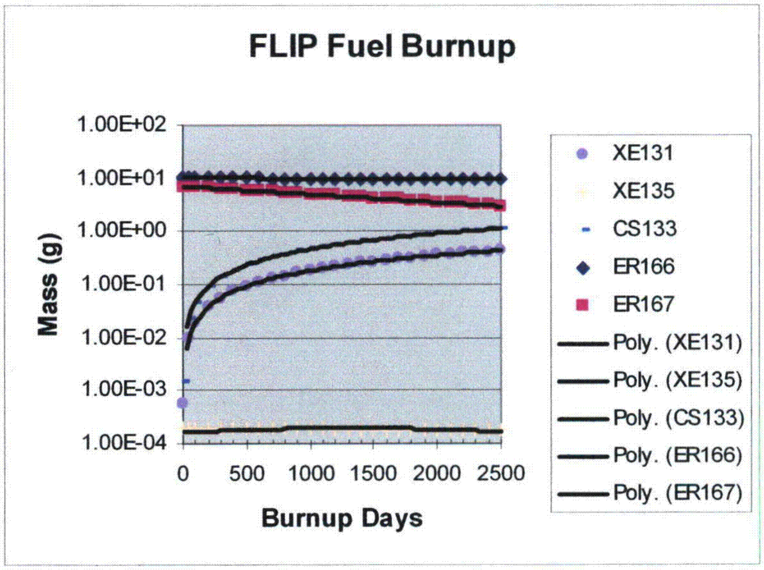

Figure 12 Masses for Typical Heavy Metals and Fission Products – Core 34A

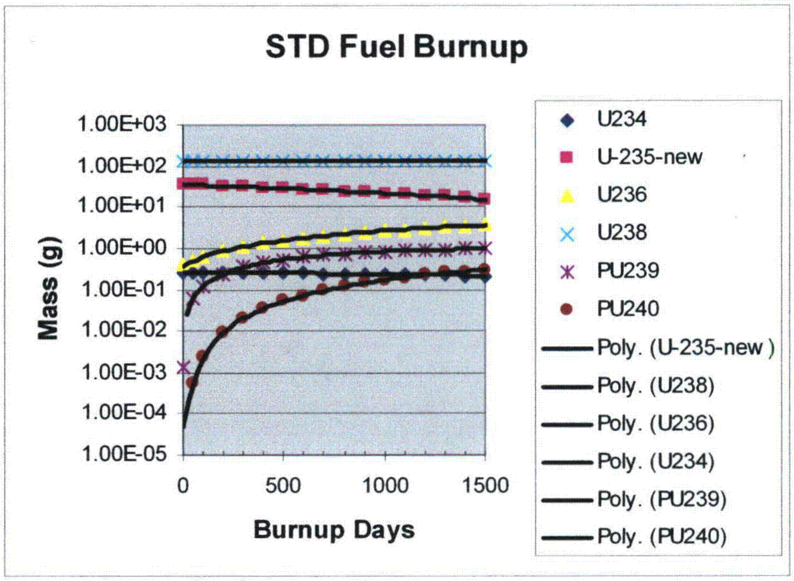

Figure 13 Masses for Typical Heavy Metals and Fission Products – Core 34A

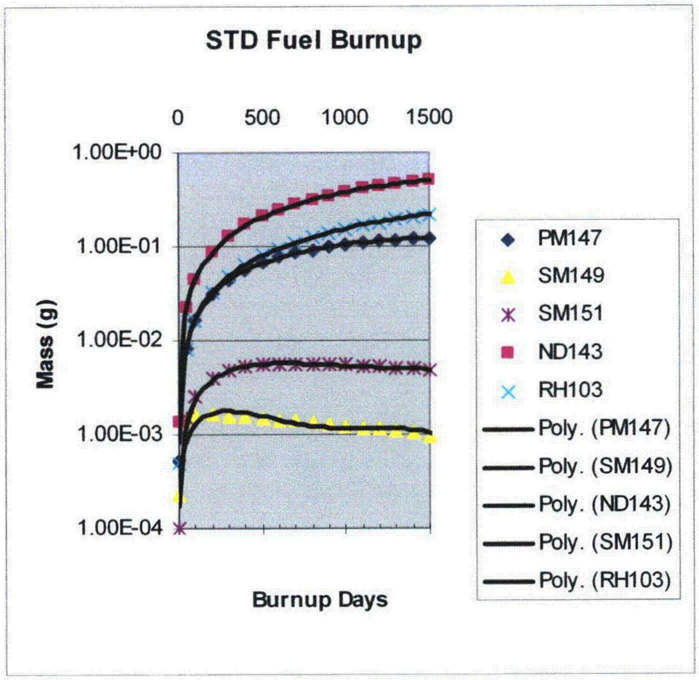

Figure 14 Masses for Typical Heavy Metals and Fission Products – Core 34A

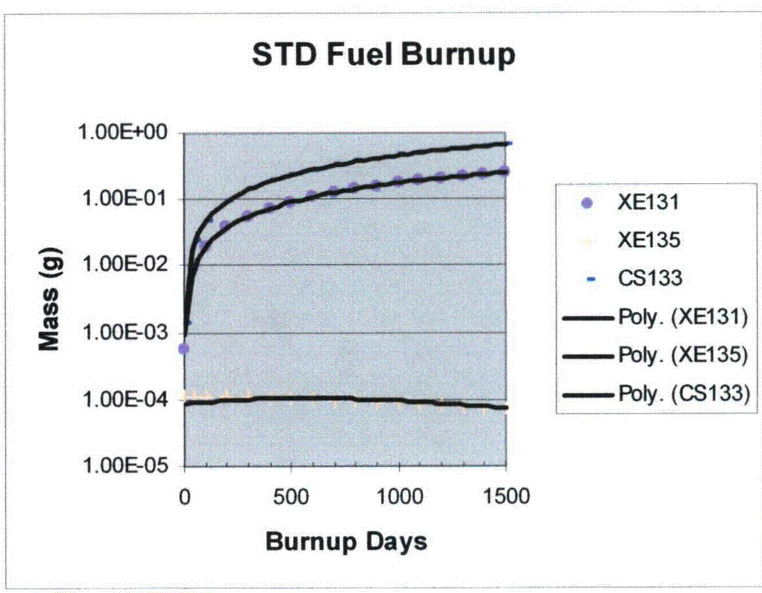

Figure 15 Masses for Typical Heavy Metals and Fission Products – Core 34A

Table 6 Nuclide densities for Burned FLIP/HEU fuel meat, MCNP Model for WSU Mixed HEU Core 34A

Nuclide	C-3	C-4	C-5	C-6	D-2
U-235	8.0007E-04	7.7473E-04	7.7462E-04	8.1192E-04	7.9579E-04
U-234	6.2968E-06	6.2201E-06	6.2197E-06	6.3325E-06	6.2838E-06
U-236	2.1441E-05	2.6122E-05	2.6144E-05	1.9233E-05	2.2236E-05
U-238	3.6591E-04	3.6499E-04	3.6498E-04	3.6634E-04	3.6576E-04
Pu-239	2.7087E-06	3.3299E-06	3.3327E-06	2.3983E-06	2.8177E-06
Pu-240	3.2430E-07	5.0162E-07	5.0250E-07	2.5159E-07	3.5226E-07
Rh-103	2.2324E-06	2.8306E-06	2.8333E-06	1.9489E-06	2.3342E-06
Xe-131	1.9948E-06	2.5322E-06	2.5347E-06	1.7405E-06	2.0861E-06
Cs-133	4.9636E-06	6.3355E-06	6.3419E-06	4.3200E-06	5.1957E-06
Nd-143	4.3222E-06	5.4824E-06	5.4877E-06	3.7725E-06	4.5196E-06
Pm-147	1.1871E-06	1.3923E-06	1.3932E-06	1.0747E-06	1.2250E-06
Sm-149	6.4244E-08	5.9000E-08	5.8988E-08	6.8133E-08	6.3001E-08
Sm-151	1.9122E-07	2.1646E-07	2.1655E-07	1.7590E-07	1.9618E-07
Er-166	1.0463E-04	1.0429E-04	1.0428E-04	1.0479E-04	1.0457E-04
Er-167	5.4872E-05	5.0590E-05	5.0571E-05	5.6972E-05	5.4129E-05
H	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02
C	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03
Zr	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02
Hf	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06
Total	9.2761E-02	9.2740E-02	9.2740E-02	9.2771E-02	9.2758E-02

Nuclide	D-3	D-4	D-5	D-6	E-3
U235	7.8678E-04	7.8611E-04	7.6130E-04	7.8202E-04	7.9388E-04
U-234	6.2566E-06	6.2546E-06	6.1793E-06	6.2422E-06	6.2781E-06
Pu-240	4.1389E-07	4.1858E-07	6.0600E-07	4.4788E-07	3.6504E-07
Rh-103	2.5476E-06	2.5632E-06	3.1435E-06	2.6597E-06	2.3796E-06
Xe-131	2.2777E-06	2.2918E-06	2.8139E-06	2.3786E-06	2.1269E-06
Cs-133	5.6840E-06	5.7200E-06	7.0613E-06	5.9417E-06	5.2994E-06
Nd-143	4.9334E-06	4.9638E-06	6.0895E-06	5.1510E-06	4.6077E-06
Pm-147	1.3004E-06	1.3057E-06	1.4838E-06	1.3379E-06	1.2415E-06
Sm-149	6.0824E-08	6.0690E-08	5.8426E-08	5.9949E-08	6.2486E-08
Sm-151	2.0566E-07	2.0631E-07	2.2621E-07	2.1018E-07	1.9830E-07
Er-166	1.0445E-04	1.0444E-04	1.0410E-04	1.0439E-04	1.0455E-04
Er-167	5.2590E-05	5.2478E-05	4.8434E-05	5.1792E-05	5.3799E-05
H	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02	5.4593E-02
C	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03	1.4961E-03
Zr	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02	3.5299E-02
Hf	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06	2.1179E-06
Total	9.2750E-02	9.2749E-02	9.2728E-02	9.2746E-02	9.2756E-02

Nuclide	E-4	E-5	E-6
U-234	6.2721E-06	6.2100E-06	6.2951E-06
U-236	2.2956E-05	2.6734E-05	2.1544E-05
U-238	3.6562E-04	3.6487E-04	3.6589E-04
Pu-239	2.9153E-06	3.4073E-06	2.7230E-06
Pu-240	3.7838E-07	5.2689E-07	3.2788E-07
Rh-103	2.4264E-06	2.9084E-06	2.2456E-06
Xe-131	2.1689E-06	2.6022E-06	2.0066E-06
Cs-133	5.4063E-06	6.5155E-06	4.9937E-06
Nd-143	4.6983E-06	5.6334E-06	4.3479E-06
Pm-147	1.2582E-06	1.4160E-06	1.1921E-06
Sm-149	6.1984E-08	5.8721E-08	6.4076E-08
Sm-151	2.0042E-07	2.1909E-07	1.9188E-07
Er-166	1.0452E-04	1.0424E-04	1.0462E-04
Er-167	5.3461E-05	5.0048E-05	5.4775E-05
H	5.4593E-02	5.4593E-02	5.4593E-02
C	1.4961E-03	1.4961E-03	1.4961E-03
Zr	3.5299E-02	3.5299E-02	3.5299E-02
Hf	2.1179E-06	2.1179E-06	2.1179E-06
Total	9.2754E-02	9.2737E-02	9.2761E-02

Table 7 Nuclide densities for Partially Burned 8.5/20 LEU fuel meat, MCNP Model for WSU Mixed HEU Core 34A and Mixed LEU Core 35A

Nuclide	*B-1*	*B-2*	*B-3*	*B-6*	**C-1**
U-235	2.0377E-04	2.1675E-04	2.0351E-04	2.0535E-04	2.1704E-04
U-234	1.8305E-06	1.8551E-06	1.8299E-06	1.8336E-06	1.8556E-06
U-236	1.1228E-05	9.1509E-06	1.1269E-05	1.0977E-05	9.1039E-06
U-238	1.0116E-03	1.0128E-03	1.0115E-03	1.0117E-03	1.0128E-03
Pu-239	3.7764E-06	2.9831E-06	3.7911E-06	3.6852E-06	2.9641E-06
Pu-240	3.5901E-07	2.1028E-07	3.6227E-07	3.3920E-07	2.0733E-07
Rh-103	1.3178E-06	9.8751E-07	1.3243E-06	1.2775E-06	9.8012E-07
Xe-131	1.1687E-06	8.7667E-07	1.1744E-06	1.1331E-06	8.7012E-07
Cs-133	2.9290E-06	2.1904E-06	2.9436E-06	2.8386E-06	2.1739E-06
Nd-143	2.4335E-06	1.8432E-06	2.4451E-06	2.3623E-06	1.8299E-06
Pm-147	7.6469E-07	6.0389E-07	7.6769E-07	7.4604E-07	6.0007E-07
Sm-149	1.7997E-08	1.9897E-08	1.7954E-08	1.8259E-08	1.9928E-08
Sm-151	6.3897E-08	5.9022E-08	6.3952E-08	6.3526E-08	5.8858E-08
H	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02
C	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03
Zr	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02
Hf	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06
Total	9.3128E-02	9.3138E-02	9.3128E-02	9.3130E-02	9.3138E-02
	C-2	C-7	D-1	D-7	E-1
U-235	1.9836E-04	2.0711E-04	2.0936E-04	2.0467E-04	2.1250E-04
U-234	1.8195E-06	1.8370E-06	1.8414E-06	1.8323E-06	1.8473E-06
U-236	1.2087E-05	1.0696E-05	1.0335E-05	1.1084E-05	9.8325E-06
U-238	1.0110E-03	1.0119E-03	1.0121E-03	1.0116E-03	1.0124E-03
Pu-239	4.0779E-06	3.5818E-06	3.4465E-06	3.7242E-06	3.2533E-06
Pu-240	4.3030E-07	3.1763E-07	2.9084E-07	3.4758E-07	2.5523E-07
Rh-103	1.4560E-06	1.2326E-06	1.1750E-06	1.2946E-06	1.0951E-06
Xe-131	1.2908E-06	1.0934E-06	1.0425E-06	1.1482E-06	9.7189E-07
Cs-133	3.2394E-06	2.7380E-06	2.6093E-06	2.8771E-06	2.4307E-06
Nd-143	2.6763E-06	2.2826E-06	2.1802E-06	2.3927E-06	2.0372E-06
Pm-147	8.2670E-07	7.2494E-07	6.9744E-07	7.5402E-07	6.5831E-07
Sm-149	1.7097E-08	1.8547E-08	1.8905E-08	1.8148E-08	1.9369E-08
Sm-151	6.4790E-08	6.3047E-08	6.2327E-08	6.3690E-08	6.1109E-08
H	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02
C	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03
Zr	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02
Hf	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06
Total	9.3125E-02	9.3131E-02	9.3132E-02	9.3129E-02	9.3135E-02

	E-2	E-7	F-1	F-2	F-3
U-235	1.9220E-04	2.0374E-04	2.1161E-04	2.0410E-04	1.9863E-04
U-234	1.8064E-06	1.8304E-06	1.8456E-06	1.8311E-06	1.8200E-06
U-236	1.3061E-05	1.1233E-05	9.9753E-06	1.1176E-05	1.2044E-05
U-238	1.0103E-03	1.0115E-03	1.0123E-03	1.0116E-03	1.0110E-03
Pu-239	4.4013E-06	3.7781E-06	3.3087E-06	3.7575E-06	4.0633E-06
Pu-240	5.1768E-07	3.5939E-07	2.6513E-07	3.5485E-07	4.2662E-07
Rh-103	1.6139E-06	1.3185E-06	1.1178E-06	1.3094E-06	1.4491E-06
Xe-131	1.4302E-06	1.1693E-06	9.9192E-07	1.1613E-06	1.2847E-06
Cs-133	3.5954E-06	2.9306E-06	2.4813E-06	2.9101E-06	3.2239E-06
Nd-143	2.9503E-06	2.4349E-06	2.0778E-06	2.4187E-06	2.6642E-06
Pm-147	8.9385E-07	7.6504E-07	6.6951E-07	7.6083E-07	8.2367E-07
Sm-149	1.6129E-08	1.7992E-08	1.9243E-08	1.8052E-08	1.7141E-08
Sm-151	6.5207E-08	6.3903E-08	6.1481E-08	6.3824E-08	6.4758E-08
H	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02	5.4712E-02
C	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03	1.4891E-03
Zr	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02	3.5684E-02
Hf	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06	2.1410E-06
Total	9.3120E-02	9.3128E-02	9.3134E-02	9.3129E-02	9.3125E-02
	F-6	F-7			
U-235	2.1347E-04	2.1105E-04			
U-234	1.8491E-06	1.8446E-06			
U-236	9.6778E-06	1.0065E-05			
U-238	1.0125E-03	1.0122E-03			
Pu-239	3.1928E-06	3.3434E-06			
Pu-240	2.4469E-07	2.7147E-07			
Rh-103	1.0707E-06	1.1321E-06			
Xe-131	9.5023E-07	1.0046E-06			
Cs-133	2.3760E-06	2.5133E-06			
Nd-143	1.9932E-06	2.1034E-06			
Pm-147	6.4611E-07	6.7654E-07			
Sm-149	1.9501E-08	1.9160E-08			
Sm-151	6.0681E-08	6.1705E-08			
H	5.4712E-02	5.4712E-02			
C	1.4891E-03	1.4891E-03			
Zr	3.5684E-02	3.5684E-02			
Hf	2.1410E-06	2.1410E-06			
Total	9.3135E-02	9.3134E-02			

Table 8 Nuclide densities for Fresh 30/20 LEU fuel meat, MCNP Model for WSU Mixed LEU Core 35A

Nuclide	Atomic Mass	Nuclide Density (atoms/b•cm)	Mass (g)
H	1.0079	0.04915763	28.86
C	12.011	0.00178701	12.50
Zr	91.224	0.03227955	1715.37
Er-166	165.93	0.00007717	7.46
Er-167	166.932	0.00005299	5.15
U-234	234.041	0.00000715	0.97
U-235	235.0439	0.00108821	149.00
U-236	236.0456	0.00000627	0.86
U-238	238.0508	0.00432194	599.33
Hf	178.49	1.93677E-06	0.20
Total		0.08877792	2519.51

Geometrical Models

The MCNP model for the full WSU Mixed HEU Core 34A consists of 51 FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs, and each fuel cluster was averaged over 4 or 3 fuel rods. The nuclide densities for each fuel cluster are shown in Tables 5, 6, and 7. The full WSU Mixed LEU Core 35A core model consists of 51 – 30/20 LEU SFEs and 68 burned 8.5/20 SFEs and each fuel cluster was averaged over 4 or 3 fuel rods. The nuclide densities for each fuel cluster are shown in Tables 6, 7, and 8.

Approach-to-Critical-WSU Mixed HEU Core 34A

Since WSU Mixed HEU Core 34A evolved over a period of time there never was a critical configuration for this core. Therefore, no MCNP model was developed for this benchmark case since there is no operational data for comparison.

Approach-to-Critical-WSU Mixed LEU Core 35A

A detailed MCNP model of the just critical WSU Mixed LEU Core 35A reactor was made which includes 47 – 30/20 LEU SFEs and 24 burned 8.5/20 SFEs, 3 blade type control rods, 1 stainless steel servo blade, 1 water-followed transient rod, and 20 graphite blocks around the core, as illustrated by Figure 16.

The critical case for the WSU Mixed LEU Core 35A was modeled to be in a position away from the thermal column (water reflected on the west side). This configuration was chosen since this is the most reactive arrangement. Figures 17 and 18 are the XY and XZ plots of the MCNP model of the Mixed LEU Core 35A cold critical case.

Figure 16 WSU Mixed LEU Core 35A – Cold Critical Core Configuration

WSU Mixed HEU Core 34A Reactor Model, Full Core (Away from Thermal Column)

A detailed MCNP model was developed for the water reflected WSU Mixed HEU Core 34A which includes 51 FLIP HEU SFEs and 68 burned 8.5/20 LEU SFEs, 3 blade type control rods, 1 stainless steel servo blade and 1 water-followed transient rod. In addition, a 2 inch thick lower cluster adapter, and a 5 inch thick aluminum grid plate below the fuel rods was modeled.

Figures 19 and 20 are the XY and XZ plots of the MCNP model of the full unrodded WSU Mixed HEU Core 34A with infinite water reflector. Figures 21 and 22 are the XY and XZ plots of the MCNP model of the full core for the WSU Mixed HEU Core 34A with all control rods inserted.

WSU Mixed LEU Core 35A Reactor Model, Full Core (Away from Thermal Column)

A detailed MCNP model was developed for the water reflected WSU Mixed LEU Core 35A, which includes 51 – 30/20 LEU SFE's, and 68 – burned 8.5/20 SFEs, 3 blade type control rods, 1 stainless steel servo blade, and 1 water-followed transient rod. In addition, it was assumed that there were 20 graphite blocks around the core, 2 inches thick lower cluster adapter, and a 5 inches thick aluminum grid plate below the fuel rods.

Figures 23 and 24 are the XY and XZ plots of the MCNP model of the full unrodded core WSU Mixed LEU Core 35A. Figures 25 and 26 are the XY and XZ plots of the MCNP model of the full core WSU Mixed LEU Core 35A with all control blades inserted.

The configuration for the Mixed LEU Core 35A is shown in Figure 27.

Figure 17 WSU Mixed LEU Core 35A, Cold Critical Configuration
Radial Model for Monte Carlo (MCNP5) Transport Calculation

Figure 18 WSU Mixed LEU Core 35A, Cold Critical Configuration
Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 19 WSU Mixed HEU Core 34A – Unrodded Full Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 20 WSU Mixed HEU Core 34A – Unrodded Full Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 21 WSU Mixed HEU Core 34A – All Rods Inserted Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 22 WSU Mixed HEU Core 34A – All Rods Inserted Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculation

Figure 23 WSU Mixed LEU Core 35A – Unrodded Full Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 24 WSU Mixed LEU Core 35A – Unrodded Full Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 25 WSU Mixed LEU Core 35A – All Rods Inserted Core Configuration Radial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 26 WSU Mixed LEU Core 35A – All Rods Inserted Core Configuration Axial Model for Monte Carlo (MCNP5) Transport Calculations

Figure 27 WSU Mixed LEU Core 35A – General Configuration

Benchmark of Mixed HEU Core

The evaluation of WSU Mixed FLIP HEU Core 34A provided an opportunity to benchmark the computational techniques to be used for evaluating the WSU Mixed LEU Core 35A.

Approach-to-Critical – WSU Mixed HEU Core 34A

Since WSU Mixed HEU Core 34A evolved over a period of time there never was a critical configuration for this core. Therefore no MCNP model was developed for this benchmark case since there is no data for comparison.

Full Unrodded Core Loading – WSU Mixed HEU Core 34A

The full core loading in the water reflected WSU Mixed HEU Core 34A contains 51 – FLIP HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The analysis for this condition included the fission product and actinides inventories calculated for each fuel cluster of Core 34A based on U-235 depletion. The MCNP calculation gave an unrodded k_{eff} value with one sigma uncertainty:

$$k_{eff} = 1.05038 \pm 0.00016$$

This is equivalent to a reactivity of \$6.31 ($\beta_{eff} = 0.0076$, see section 4.5.5). The experimentally determined value was \$6.65, based on measured control rods worth.

Full Core Loading, All Control Rods Inserted – WSU Mixed HEU Core 34A

The full core loading in the water reflected WSU Mixed HEU Core 34A contains 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation with all control rods inserted gave a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.95010 \pm 0.00016$$

This is equivalent to -$6.91 of reactivity shutdown. The control system consisting of 3 control blades, a regulating servo blade and a transient rod has a calculated worth of $13.22. It may be useful to note that the total worth of the experimentally determined individual five control rods was $12.53.

4.5.2 Critical Core Configuration; Excess Reactivity

The number of fuel rods in the WSU Mixed HEU Core 34A is 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The proposed core loading for the WSU Mixed LEU Core 35A will be 51 – 30/20 LEU SFEs and 68 burned 8.5/20 LEU SFEs.

Full Unrodded Core Loading – WSU Mixed HEU Core 34A

The full core loading for the WSU Mixed HEU Core 34A contains 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation gave an unrodded k_{eff} value with one sigma uncertainty:

$$k_{eff} = 1.05038 \pm 0.00016$$

This is equivalent to a reactivity of $6.31. The experimentally determined measured value was $6.65.

Full, Unrodded Core Loading – WSU Mixed LEU Core 35A

The full core loading in the WSU Mixed LEU Core 35A contains 51 – 30/20 LEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation gives an unrodded k_{eff} value with one sigma uncertainty of:

$$k_{eff} = 1.05019 \pm 0.00017$$

This corresponds to a core reactivity of $6.37 ($\beta_{eff} = 0.0075$, see section 4.5.5).

4.5.3 Worth of Control Rods

Full Core Loading, All Control Rods Inserted – WSU Mixed HEU Core 34A

The full core loading for the water reflected WSU Mixed HEU Core 34A contains 51 – FLIP/HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation with all control rods inserted gave a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.95010 \pm 0.00016$$

This is equivalent to a reactivity shutdown of -$6.91. The five control rods have a calculated worth of $13.22.

The individual control rod/blade's worths (calculated and measured) are given in Table 9. The discrepancy between the sum of the calculated individual rod/blade worths, as shown in Table 9 ($10.68), and the calculated worth with all rods/blades out and all rods/blades in ($13.22) is due to the shadowing effects of adjacent control rods/blades when worths are calculated individually.

Table 9 WSU Mixed HEU Core 34A – Control Rod Worth

	Calculated MCNP	Measured
Blade 1 (Shim)	$ 1.32 ± 0.03	$ 1.68
Blade 2 (Shim)	$ 2.89 ± 0.03	$ 3.56
Transient Rod 3	$ 3.22 ± 0.03	$ 3.11
Blade 4 (Shim)	$ 2.86 ± 0.03	$ 3.99
Blade 5 (Servo)	$ 0.40 ± 0.03	$ 0.19
Total	$ 10.68 ± 0.07	$ 12.53

Full Core Loading, All Control Rods Inserted – WSU Mixed LEU Core 35A

The full core loading in the WSU Mixed LEU Core 35A reactor contains 51 – 30/20 LEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The MCNP calculation with all control rods inserted gives a k_{eff} value with one sigma uncertainty:

$$k_{eff} = 0.94717 \pm 0.00017$$

This is equivalent to reactivity shutdown of -$7.44. The calculated six control rods have a combined reactivity worth of $10.97.

The individual calculated control rod/blade's worths are given in Table 10.

Table 10 WSU Mixed LEU Core 35A – Control Rod Worth

	Calculated MCNP
Blade 1 (Shim)	$ 1.34 ± 0.03
Blade 2 (Shim)	$ 2.99 ± 0.03
Transient Rod 3	$ 3.19 ± 0.03
Blade 4 (Shim)	$ 3.02 ± 0.03
Blade 5 (Shim)	$ 0.43 ± 0.03
Total	$ 10.97 ± 0.07

4.5.4 Shutdown Margin for Mixed HEU and Mixed LEU Cores

Shutdown Margin, WSU Mixed HEU Core 34A

As stated in the Technical Specifications, the reactor shall not be operated unless the shutdown margin provided by the control rods is greater than $0.25 with:

a) The highest worth non-secured experiment in its most reactive state,

b) The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and

c) The reactor in the cold condition without xenon.

The MCNP code has been used to evaluate the individual worth of the five control rods. The control rod worths are shown in Table 9. The transient rod is calculated to be most reactive rod at $3.22. Actual measurements have shown that shim blade 4 is the highest worth rod with a worth of $3.99. The calculated shutdown margin (the core excess of $6.31 minus the total worths of blades 1, 2 and 3) is -$0.76 and the measured shutdown margin (the core excess of $6.65 minus the total worths of blades 1, 2 and transient rod) is -$1.70. Both this calculated and measured give adequate shutdown margin for Core 34A.

Shutdown Margin, WSU Mixed LEU Core 35A

As stated in the applicable Technical Specifications, the reactor shall not be operated unless the shutdown margin provided by control rods is greater than $0.25 with:

a) The highest worth non-secured experiment in its most reactive state,

b) The highest worth control rod and the regulating rod (if not scrammable) fully withdrawn, and

c) The reactor in the cold condition without xenon.

The MCNP 5 code was run for the WSU Mixed LEU Core 35A with the most reactive rod plus the non-scrammable regulating servo blade up out of the core and the three shim blades inserted in the core. The individual control rod's worth for Core 35A are given in Table 10. The transient rod is calculated to be the most reactive rod at $3.19. The calculated shutdown margin (the core excess of $6.37 minus the total worths of blades 1, 2 and 3) is -$0.98 which gives an adequate shutdown margin for WSU Mixed LEU Core 35A.

4.5.5 Additional Core Physics Parameters for Mixed HEU and Mixed LEU Cores

Effective Delayed Neutron Fraction, βeff – WSU Mixed HEU Core 34A

The effective delayed neutron fraction, β_{eff}, was derived from Monte Carlo calculations of the WSU Mixed HEU Core 34A with all control rods out.

The computed values for K_t and K_p are used in the following expression to obtain β_{eff}

$\beta_{eff} = 1 - [K_p / K_t]$

Where: K_p = core reactivity using prompt fission spectrum
K_t = core reactivity using prompt and delayed fission spectrum

The values of K_p and K_t calculated using MCNP are:

$K_p = 1.04244 \pm 0.00016$
$K_t = 1.05038 \pm 0.00016$

Using these values the result for WSU Mixed HEU Core 34A is:

$$\beta_{eff} = 0.0076 \pm 0.0002$$

Effective Delayed Neutron Fraction, βeff – WSU Mixed LEU Core 35A

The effective delayed neutron fraction, β_{eff}, for the WSU Mixed LEU Core 35A is calculated exactly as for the WSU Mixed HEU Core 34A above but with the updated WSU Mixed LEU Core 35A input parameters.

The values of K_p and K_t calculated using MCNP are:

$K_p = 1.04225 \pm 0.00017$
$K_t = 1.05019 \pm 0.00017$

The result for the Mixed TRIGA LEU fuel is:

$$\beta eff = 0.0075 \pm 0.0002$$

Prompt Neutron Life (ℓ) – WSU Mixed HEU Core 34A

The prompt neutron lifetime, ℓ, was computed by the 1/v absorber method where a very small amount of boron is distributed homogeneously throughout the system and the resulting change in reactivity is related to the neutron lifetime. This calculation was done using the 3-D diffusion theory model for the core to allow very tight convergence of the problems. The boron cross sections used in the core were generated over a homogenized core spectrum. Boron cross sections used in all other zones were generated over a water spectrum.

The neutron lifetime, ℓ, is defined as follows:

$$\ell = \Delta k_{eff}/\omega$$

where Δk_{eff} is the change in reactivity due to the addition of boron and ω is related to the boron atom density and,

$$N_B = \omega/\delta_o\, v_o = 6.0205 \times 10^{-7}$$

where N_B = boron density (atoms/b•cm)
ω = integer = 100 (the calculation is insensitive to changes in ω between 1 and 100)
v_o = 2200 m/sec
δ_o = 755 barns = δ_a^B at 2200 m/sec

As described in the β_{eff} section above, the 3-D model used very tight convergence criteria (1.0×10^{-8} of k_{eff}, 1.0×10^{-6} point flux). The cases were run cold (23°C) with fresh FLIP fuel. The result for prompt neutron lifetime in the unrodded core is:

$$\ell = 30.7\ \mu\text{sec}$$

Prompt Neutron Life (ℓ) – WSU Mixed LEU Core 35A

Using the same 1/v absorber method described above for the WSU Mixed HEU Core 34A, the prompt neutron life (ℓ) has been evaluated for the WSU Mixed LEU TRIGA fuel in Core 35A.

The result for the prompt neutron life (ℓ) in the unrodded WSU Mixed LEU Core 35A is:

$$\ell = 28.2\ \mu\text{sec}$$

Prompt Negative Temperature Coefficient of Reactivity, α-WSU Mixed HEU Core 34A

The definition of α, the prompt negative temperature coefficient of reactivity, is given as

$$\alpha = \frac{d\rho}{dT}$$

where ρ = reactivity
= (k-1)/k
T = reactor temperature (°C)

$$\alpha = \frac{1}{k^2}\frac{dk}{dT}$$

To evaluate ($\Delta\rho$) from reactivity as a function of reactor core temperature, the finite differences can be written as follows:

$$\Delta\rho_{1,2} = \frac{k_2 - 1}{k_2} - \frac{k_1 - 1}{k_1}$$

$$= \frac{k_2 - k_1}{k_1 k_2}$$

Thus, $$\alpha_{1,2} \cong \frac{k_2 - k_1}{k_1 k_2} \text{ x } \frac{1}{\Delta T_{1,2}}$$

The data in Table 11 were produced by DIF3D for the listed core temperatures.

Figure 28 is a histogram plot of the computed values of α as a function of reactor temperature.

Table 11 Reactivity Change with Temperature – WSU Mixed HEU Core 34A

Average Core Temperature °C	**k_{eff}**	**Δ k_{eff}**	**$\frac{k_a - k_b}{k_a k_b}$**	**$\alpha_{a,b}$ (Δk/k per °C)**
23	1.04196			
		0.01023	0.009516	5.376E-05
200	1.03173			
		0.00659	0.006231	7.788E-05
280	1.02514			
		0.01181	0.011369	9.474E-05
400	1.01333			
		0.0369	0.037294	1.243E-04
700	0.97643			
		0.04137	0.045311	1.510E-04
1000	0.93506			

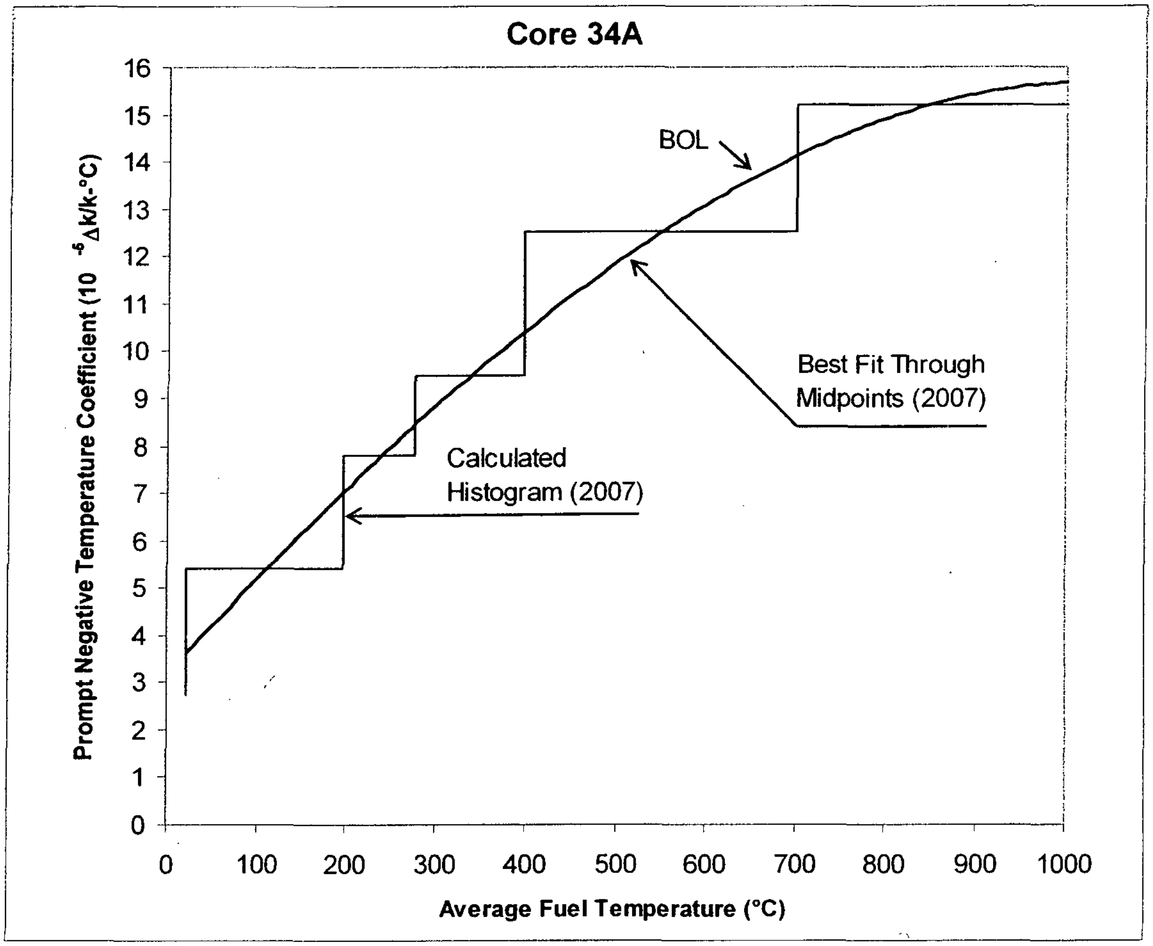

Figure 28 Prompt Negative Temperature Coefficient vs. Fuel Temperature – WSU Mixed HEU Core 34A.

Prompt Negative Temperature Coefficient of Reactivity, α – WSU Mixed LEU Core 35A

Following the procedure set forth in the section for the WSU Mixed HEU Core 34A, the calculated results from DIF3D for the WSU Mixed LEU Core 35A are listed in Table 12 for Beginning-of-Life (BOL) and in Table 13 for End-of-Life (EOL).

Figure 29 is a histogram plot of the computed values for α in Tables 12 and 13 as a function of core temperature for both BOL and EOL.

In Figure 29, it can be seen that the prompt negative temperature coefficient (α) for WSU Mixed LEU Core 35A has only a modest decrease in values at 1000 MWD burnup (EOL) (e.g., 12.5 x 10^{-5} to 10.5 x 10^{-5} Δk/k-°C at 700-1000°C).

Table 12 Reactivity Change with Temperature, WSU Mixed LEU Core 35A, (BOL)

Average Core Temperature °C	k_{eff}	Δk_{eff}	$\frac{k_a - k_b}{k_a k_b}$	$\alpha_{a,b}$ ($\Delta k/k$ - °C)
23	1.03863			
		0.0113	0.010590	5.983E-05
200	1.02733			
		0.00646	0.006160	7.699E-05
280	1.02087			
		0.01101	0.010680	8.900E-05
400	1.00986			
		0.03237	0.032792	1.093E-04
700	0.97749			
		0.03506	0.038058	1.269E-04
1000	0.94243			

Table 13 Reactivity Change with Temperature, WSU Mixed LEU Core 35A, (EOL)

Average Core Temperature °C	k_{eff}	Δk_{eff}	$\frac{k_a - k_b}{k_a k_b}$	$\alpha_{a,b}$ ($\Delta k/k$ - °C)
23	1.00916			
		0.01018	0.010098	5.705E-05
200	0.99898			
		0.00556	0.005603	7.003E-05
280	0.99342			
		0.00923	0.009440	7.867E-05
400	0.98419			
		0.02626	0.027854	9.285E-05
700	0.95793			
		0.02841	0.031906	1.064E-04
1000	0.92952			

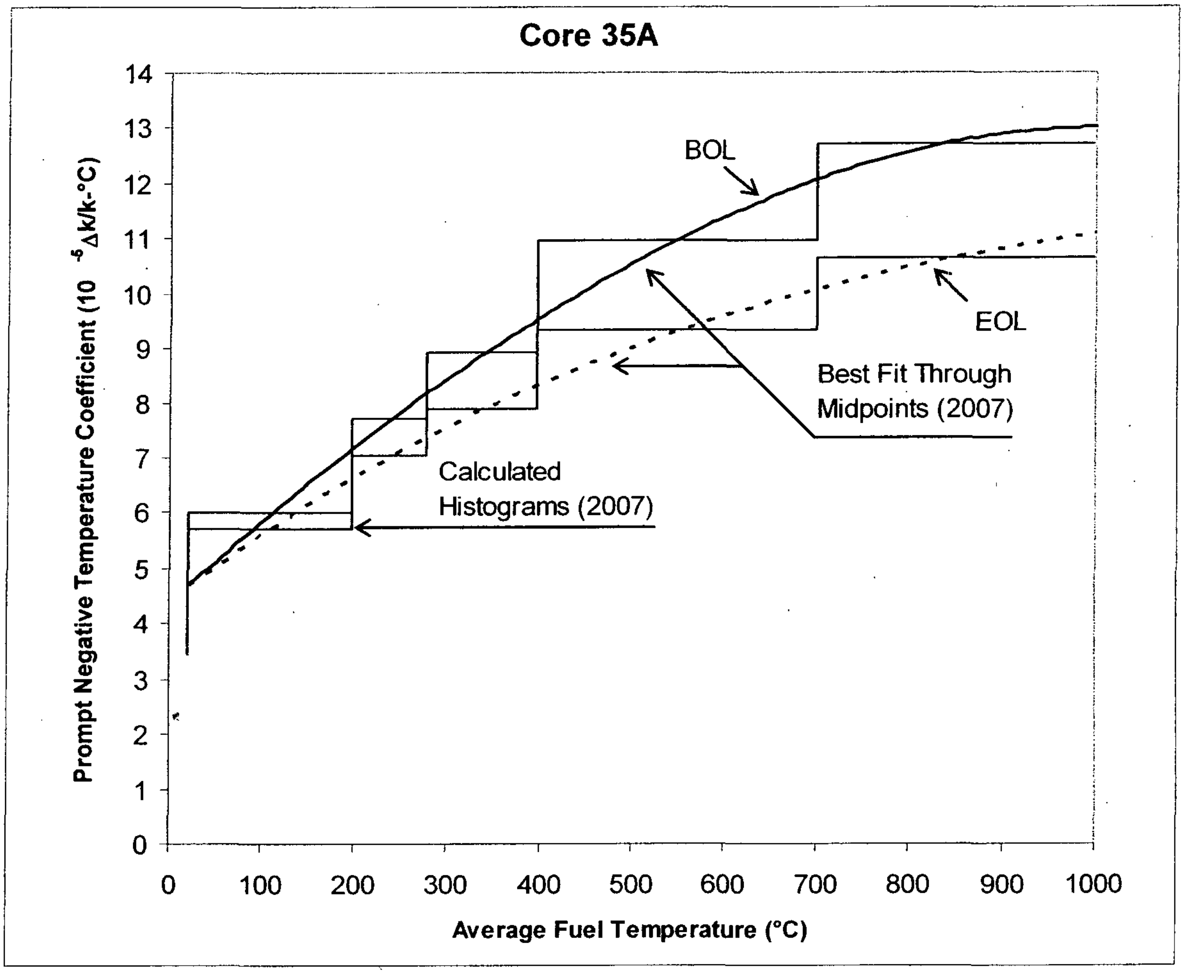

Figure 29 Prompt Negative Temperature Coefficient – WSU Mixed LEU Core 35A Fuel, Beginning of Life (BOL) and End of Life (EOL)

Void Coefficient – WSU Mixed HEU Core 34A

The "void" coefficient of reactivity is defined for a TRIGA reactor as the negative reactivity per 1% void in the reactor core water. For the WSU Mixed HEU Core 34A reactor, the calculated void coefficient is about 0.080% Δk/k per 1% water void (negative reactivity). This void coefficient is not normally considered a safety concern for TRIGA reactors. The reason is the relatively small size of this coefficient and the fact that all TRIGA reactors are significantly undermoderated. Therefore, if a portion of the core water is replaced with a low density material (i.e., steam, gas including air, etc.), a negative reactivity will occur. An example would be placing a dry, experimental tube, with a void volume of 205 cc in the 38.1 cm core fueled height, in the central region of the core (replacing a fuel rod) and then being accidentally flooded with water. The calculated loss in core reactivity would be about \$ 0.10 ± 0.03. A safety effect of rapid reactivity insertion to be considered is the effect of accidental flooding of an in-core dry experimental tube such as postulated above. In this case, the rapid reactivity insertion would be only about \$ 0.10. The insertion of \$ 0.10 reactivity is far less than \$1.00 (prompt critical).

The conclusion is that the very small void coefficient of reactivity is not a source of safety concern.

Void Coefficient – WSU Mixed LEU Core 35A

The void coefficient for the WSU Mixed LEU Core 35A is 0.135% $\Delta k/k$ per 1% water void (negative reactivity). Using the same example from above in placing a dry, experimental tube, with a void volume of 205 cc in the 38.1 cm core fueled height, in the central region of the core (replacing a fuel rod) and then being accidentally flooded with water, the prompt gain in reactivity is about \$ 0.18 ± 0.03. This reactivity addition is far less than \$1.00 required for prompt critical.

The conclusion is that the very small void coefficient is not a source of safety concern.

4.5.6 Core Burnup – WSU Mixed LEU Core 35A

Burnup analyses were performed using the DIF3D multi-dimensional diffusion theory code along with the BURP depletion code. All burnup analyses used the cross-sections generated for Beginning-of-Life (BOL) concentrations at the approximate average fuel temperature of 280°C, the closest nuclear data available.

Figure 30 shows the results from design calculations for core excess reactivity as a function of burnup. The time steps used for the burnup calculation started with 3 days (to evaluate equilibrium xenon poisoning) and then 50 day intervals from time 0 at full power (1.0 MW). The LEU burnup curve in Figure 30 gives a lifetime of the initial core (with no fuel shuffling) of about 1000 MWD at 1.0 MW, full equilibrium xenon poisoning, and about \$ 0.40 reactivity left for burnup or experiments. For comparison, a burnup curve for WSU Mixed HEU Core 34A was calculated and is also shown in Figure 30. The WSU Mixed HEU Core 34A burnup curve gives a lifetime of the initial core (with no fuel shuffling) of about 1000 MWD at 1.0 MW, full equilibrium xenon poisoning, and about \$0.67 reactivity remaining.

The data on burnup at the 3-day interval indicates a xenon equilibrium poison value of about \$1.38. Xenon is produced interior to the TRIGA fuel elements where the thermal neutron flux is severely depressed due to 30 wt. % uranium and erbium burnable poison. At the end of core life, an independent calculation gives an equilibrium xenon poison value of \$1.59. This value is larger at EOL because the thermal flux in the fuel is larger due to burnup of a portion of the U-235 and erbium.

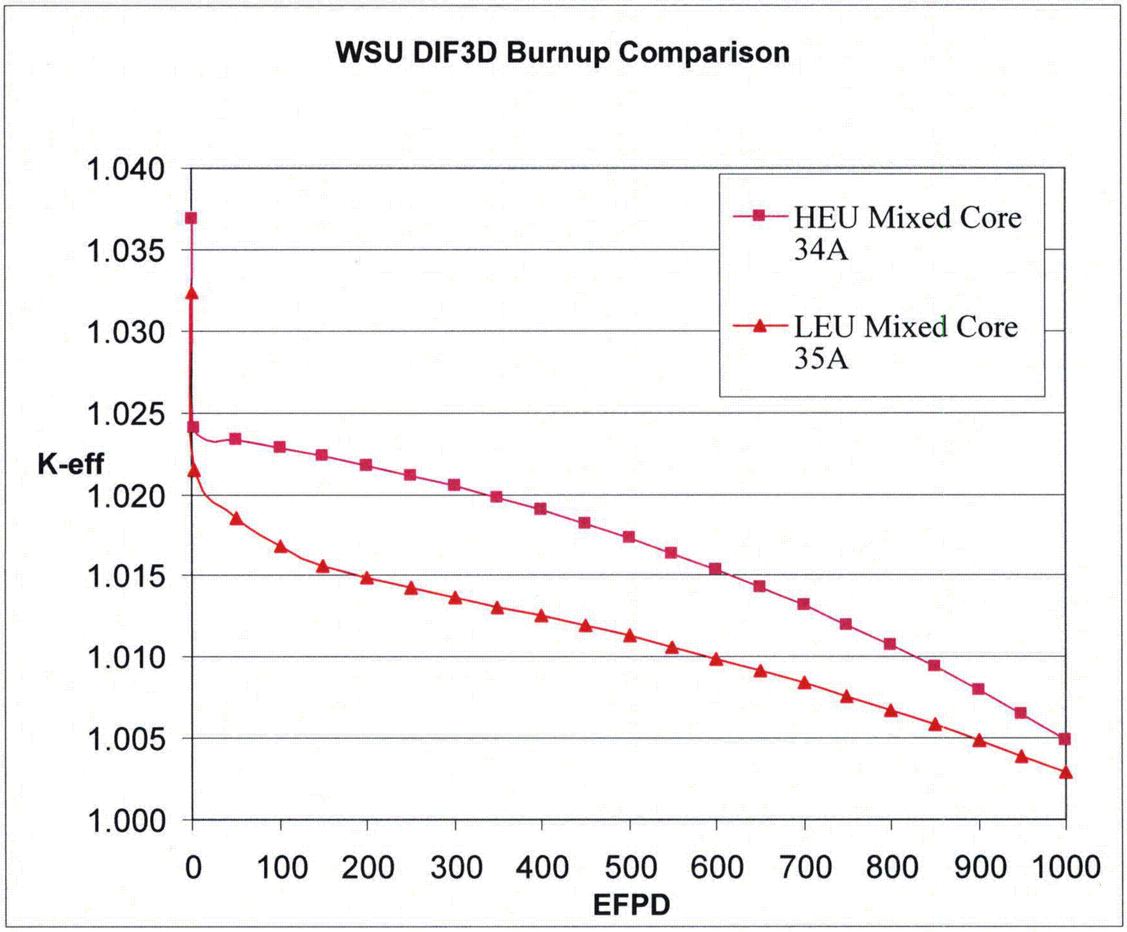

Figure 30 Reactivity versus Burnup for 1 MW TRIGA HEU Mixed and LEU Mixed (30/20) Cores

Reactor Parameters at 1000 MWD Burnup

Using the procedure set forth above for the delayed neutron fraction, the effective delayed neutron fraction has been evaluated for the LEU (30/20) core at 1000 MWD burnup. The value obtained is

$$\beta = 0.0073 \pm 0.0002,$$

which is only slightly less than the beginning of core life value of 0.0075.

Similarly, using the procedure set forth above for the prompt neutron life, the prompt neutron life (ℓ) at 1000 MWD burnup has been evaluated for the LEU (30/20) core. The value obtained is

$$\ell = 27.0 \ \mu\text{sec}$$

which is slightly less than the beginning of core life value of 28.2 μsec.

The values of the prompt negative temperature coefficient for TRIGA LEU (30/20) fuel at 1000 MWD burnup have already been presented in Tables 12 and 13 and shown in Figure 29 compared with the values at beginning of life.

4.5.7 Reactivity Loss at Reactor Power

Reactivity Loss, WSU Mixed HEU Core 34A

The prompt negative temperature coefficient of reactivity is active in all reactor operations for which the fuel temperature is elevated above ambient temperature. Consequently, core reactivity is lost at any power above a few kilowatts (when fuel temperatures begin to rise). Calculations of k-effective have been made for reactor powers of 0.5, 1.0, and 1.3 MW, respectively. During the operation of WSU Mixed HEU Core 34A, some measurements were made at 1.0 MW to evaluate the reactivity losses. The calculated and measured values of reactivity losses are tabulated in Table 14.

Table 14 Calculated and Measured Reactivity Loss, WSU Mixed HEU Core 34A

P(MW)	$ (a)	$ (b)	$\Delta\rho$ ($)$_{calc}$ (a)	$\Delta\rho$ ($)$_{calc}$ (b)	$\Delta\rho$ ($)$_{meas}$
0	5.299	6.31			
0.5	4.431	5.28	0.87	1.03	
1.0	3.845	4.58	1.45	1.73	2.20
1.3	3.716	4.42	1.58	1.88	

(a) calculated by DIF3D
(b) DIF3D results normalized to MCNP results

As can be seen from a comparison of measured and calculated reactivity loss, the measured reactivity loss at 1.0 MW is greater than the calculated value. The magnitude of the reactivity losses are related directly to the calculated temperatures. In Table 27, the measured temperatures ($T_{0.3}$) agree well with the calculated $T_{0.3}$ temperatures at 1.0 MW. Thus, the lack of agreement for reactivity losses could be attributed to a larger gap in low power elements resulting in higher average core temperature. If the core average temperature is 290°C instead of 221°C, Table 26, then the reactivity loss is calculated to be $2.18.

Reactivity Loss, WSU Mixed LEU Core 35A

Calculations of core reactivity were made for operating power levels up to 1.3 MW. These calculated values of the loss in reactivity are tabulated in Table 15. The reactivity loss at 1.0 MW is $2.52 (cold-hot reactivity swing). After burnup to 1000 MWD, the reactivity loss at 1.0 MW is predicted to drop to $1.60 partly due to a lower core average temperature at EOL.

Table 15 Calculated Reactivity Loss, WSU Mixed LEU Core 35A

P(MW)	$ (a)	$ (b)	$\Delta\rho$ ($) (a)	$\Delta\rho$ ($) (b)
0	4.959	6.37	0	
0.5	3.129	4.019	1.83	2.35
1.0	2.436	3.129	2.52	3.24
1.3	2.194	2.818	2.76	3.55

(a) calculated by DIF3D
(b) DIF3D results normalized to MCNP results

4.5.8 Power Peaking; WSU Mixed HEU Core 34A

Power peaking in the core is analyzed on the basis of the following component values:

1. $\overline{P}_{rod} / \overline{P}_{core}$: Rod Power Factor (RPF) – The power generation in a fuel rod (element) relative to the core averaged rod power generation.

2. $(\hat{P}/\overline{P})_{axial}$: Axial Power Factor (APF) – Axial peak-to-average power ratio within a fuel rod.

3. $(\hat{P}_{rod} / \overline{P}_{rod})_{radial}$: Intra Rod peaking factor (Intra Rod), the peak-to-average power in a radial plane within a fuel rod.

Since maximum fuel temperature is the limiting operational parameter for the core, the peaking factor of greatest importance for steady-state operation is the RPF. The maximum value of this factor for the hottest rod, the hot-rod factor, determines the power generation in the hottest fuel element. When combined with the axial power distribution, the hot-rod factor is used in the thermal analysis for determination of the maximum fuel temperature. The radial power distribution within the element has only a small effect on the peak temperature but is also used in the steady-state thermal analysis.

The Intra Rod peaking factor is of importance in the transient analysis for calculating maximum fuel temperatures in the time range where the heat transfer is not yet significant. It is used in the safety analysis to calculate the peak fuel temperature under adiabatic conditions, where temperature distribution is the same as power distribution.

Peaking factors calculated for the WSU Mixed HEU Core 34A are shown in Table 16 and the axial power distribution is shown in Figure 31. Values are shown for the Hot Rod (D4NE), Average Rod, and the two IFE fuel Elements located in positions D6NW and C4NW.

Fuel temperatures for selected reactor power levels have also been calculated for the hottest and average fuel rods. These results are presented in Table 27 together with the experimentally measured fuel temperature for 1.0 MW.

It will be noted that the instrumented fuel elements are located in core locations D6NWand C4NW, these locations are not the hottest core locations. Since the sensing tip of the thermocouple is 0.30 inches from the axial centerline of the fuel element, the temperatures reported in Table 26 are calculated for the hottest radial position ($\hat{T}$) and for a position 0.30 inches from the center line ($\hat{T}_{0.3}$). Finally, the average core temperature ($\overline{T}_{avg\ core}$) was calculated.

Table 16 Power Peaking Factors – WSU Mixed HEU Core 34A

Current Operating Condition – Rods at Critical Positions						
	Cold Critical - 23°C			Hot Critical - 280°C		
	RPF	APF	Intra-Rod	RPF	APF	Intra-Rod
Hot Rod	2.56	1.27	2.01	2.49	1.29	1.99
Ave Rod	1.00	1.27	1.50	1.00	1.33	1.50
IFE 1 – D6NW	1.77	1.27	0.85	1.69	1.28	0.85
IFE 2 – C4NW	1.39	1.13	0.85	1.51	1.44	0.85

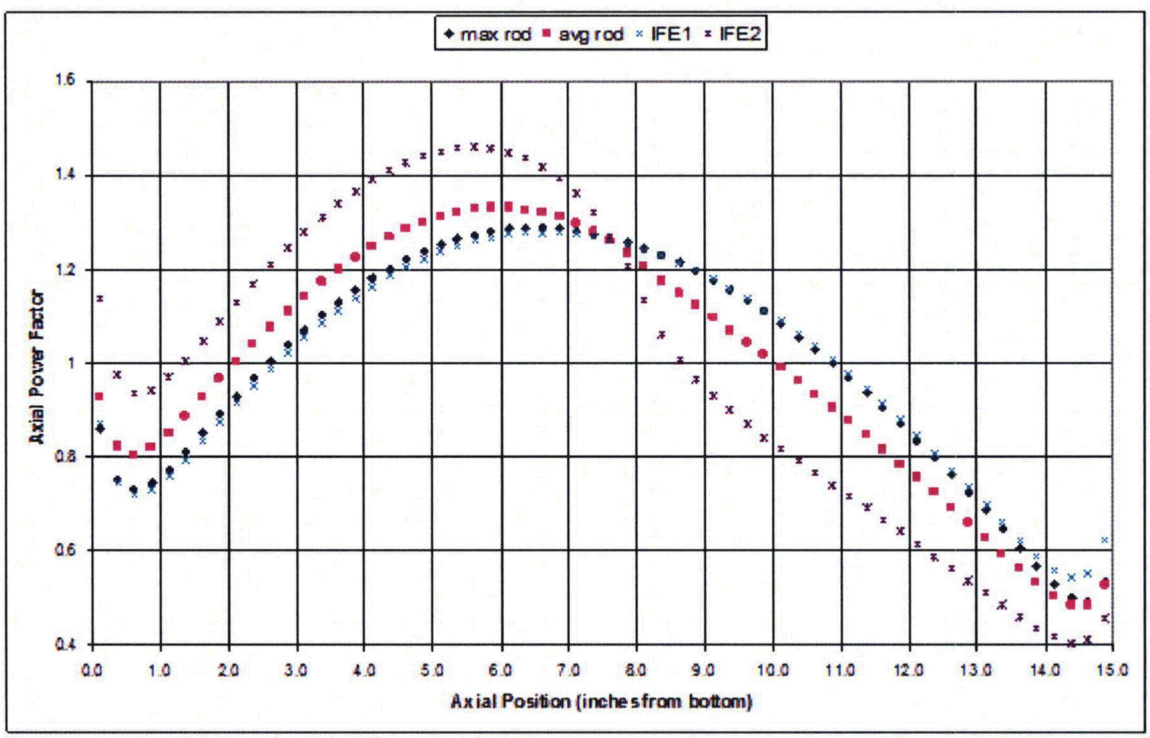

Figure 31 WSU Mixed HEU Core 34A – Axial Power Profile versus Distance from Bottom of Fueled Section

4.5.9 Power Peaking; WSU Mixed LEU Core 35A

The peaking factors calculated for the WSU Mixed LEU Core 35A at both BOL and EOL are shown in Table 17 and 18. The axial power distributions are shown in Figure 32.

Table 17 Power Peaking Factors – WSU Mixed LEU Core 35A - BOL

Beginning of Life _ Rods at Critical Positions						
	Cold Critical - 23°C			**Hot Critical - 280°C**		
	RPF	APF	Intra-Rod	RPF	APF	Intra-Rod
Hot Rod	2.56	1.27	1.35	2.47	1.29	1.19
Ave Rod	1.00	1.29	1.55	1.00	1.33	1.55
IFE 1 – D6NW	1.73	1.26	0.51	1.64	1.27	0.45
IFE 2 – C4NW	1.43	1.17	0.51	1.56	1.44	0.45

Table 18 Power Peaking Factors – WSU Mixed LEU Core 35A – All rods out, EOL

End of Life – All Rods Out			
	Hot Critical - 280°C		
	RPF	APF	Intra-Rod
Hot Rod	2.33	1.27	1.29
Ave Rod	1.00	1.25	1.52
IFE 1 – D6NW	1.55	1.26	0.49
IFE 2 – C4NW	1.78	1.24	0.49

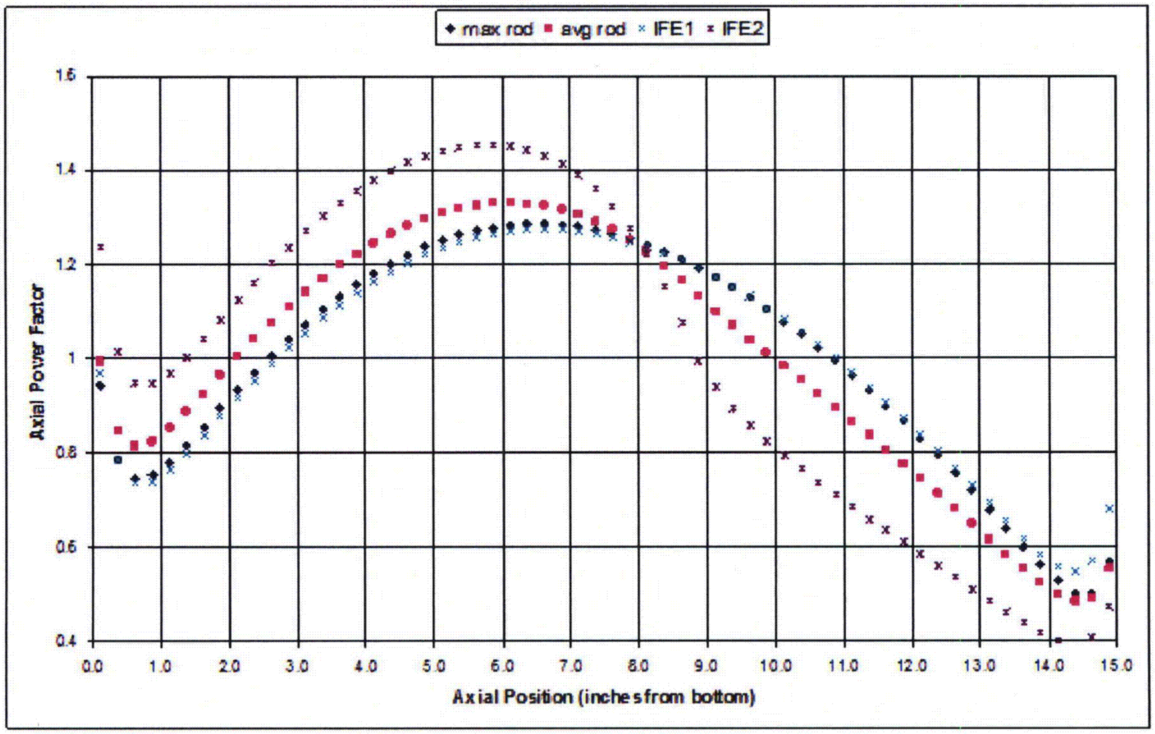

Figure 32 WSU Mixed LEU Core 35A – Axial Power Profile versus Distance from Bottom of Fueled Section

4.5.10 Pulsing Operation, WSU Mixed HEU Core 34A

Most of the 65 TRIGA reactors have pulsing capability. Thousands of TRIGA reactor pulses have been safely performed. A calculation procedure (TRIGA-BLOOST) based on a space-independent kinetics model, Ref. 8, has been developed for predicting pulse performance of TRIGA reactors.

The following simplified relationships are given to show qualitatively how the pulsing performance is influenced by the important reactor parameters:

$$\tau = \ell / \Delta k_p = \text{reactor period}$$

$$\overline{\Delta T} = \frac{2\Delta k_p}{\alpha} = E / C$$

$$\hat{P} = \frac{C(\Delta k_p)^2}{2\alpha \ell} = \text{peak pulsed power}$$

$$E = \frac{2C\Delta k_p}{\alpha} = \text{total energy release in prompt burst}$$

where

ℓ = prompt neutron life

α = prompt negative temperature coefficient
C = total heat capacity of the core available to the prompt pulse energy release
$\overline{\Delta T}$ = change in average core temperature produced by the prompt pulse
Δk_p = that portion of the step reactivity insertion which is above prompt critical

Water filled regions within the core promote flux peaking and result in increased power peaking and peak fuel temperatures, especially during a reactivity pulse. The WSU Mixed HEU Core 34A is a compact core with no in-core experimental regions that could be water filled. However, the transient rod (Rod 3) is water-followed and thus constitutes a region of enhanced power peaking. This region is correctly modeled in the codes that are used.

The BLOOST pulsing performance results have been prepared for the currently operating WSU Mixed HEU Core 34A and are shown in Table 19 for reactivity insertions of \$1.50, \$1.75, \$2.00, \$2.30, \$2.50. Also shown in Table 19 is the peak fuel temperature in the core, fuel element (D4NE), and where available, the measured temperature results ($\hat{T}_{0.3}$) for the instrumented fuel elements in positions D6NW and C4NW are also shown.

Figure 33 shows a plot of the pulsed fuel temperatures as a function of reactivity insertion calculated using an assumed fuel to cladding gap of 0.2 mils. Since the peak pulsed fuel temperature will be limited to 830°C during a pulse, the reactivity insertion (\$2.02) to produce this temperature is also indicated in Figure 33.

Measured data from pulses performed at various times during the operation of Core 34A are included in Tables 19 and 20. These data are also presented in Tables 21 and 22. Table 21 illustrates the Calculated Current Pulse Performance for WSU Mixed HEU Core 34A, along with the Calculated Pulse Performance for Core 35A, at BOL. Numerous test pulses have been performed on the WSU Mixed HEU core 34A, while also recording relevant data parameters as functions of reactivity added, including peak temperatures for the two IFE's (D6NW and C4NW), energy release, and peak pulse power. Table 20 provides pulsing data for the WSU Mixed core. There are six \$2.00 pulses, and three \$2.15 pulses included in the table. Calculations performed at WSU have indicated that the reactor may be safely pulsed to \$2.20 with core configuration 34A. The data provided in Table 19A include results of direct measurements of energy release, as indicated on the Nuclear Power Pulse Channel, NPP-1000. The functionality of the NPP-1000 channel is confirmed by comparison with the Nuclear Multi-Range Linear channel, NMP-1000, during routine operations. Comparison of the actual measured energy release with the modeled values shows that the calculated values consistently over-predict the pulse energy release. As a result, the BLOOST code output for core 34A is very conservative. Since the same code and procedures were used to develop the model for core 35A, it is reasonable to conclude that the model for core 35A is also very conservative.

Initially, WSU will limit pulsing in core 35A to $2.02, as described in this SAR. WSU will conduct a series of test pulses with reactivity values less than $2.02, measuring the pulsing parameters, including peak power and total energy release, to experimentally determine a safe pulsing limit for core 35A.

The pulse is completed about 0.3 seconds after pulse initiation, at which time the peak fuel temperature is computed. The energy results from the BLOOST-calculations are reported at a time of about 1 second after the pulse initiation (well after the peak pulsed power) and at about the time a control rod SCRAM is initiated. Thereafter, the peak fuel temperature (at the outer surface of the fuel rods) decreases as energy flows both to the center of the fuel rod and to the cooling water. However, the average core temperature continues to rise as energy is accumulated from the "tail" of the pulse until the pulse control rod scrams, shutting down the pulsed reactor.

Account has been taken for the power peaking in the water filled region when the transient rod is pulsed. The hottest fuel rod, in core position D4NE, is adjacent to the transient rod, reflecting this power peaking.

Table 19 Pulse performance: Measured and Calculated, WSU Mixed HEU Core 34A

Parameter	Pulse				
	$1.50	**$1.75**	**$2.00**	**$2.30**	**$ 2.50**
Measured Data (a)					
$\hat{P}$ (MW)	240	440	1030		
E(MW – sec)	16	20	25	N/A	N/A
$\hat{T}_{0.3}$ (°C)					
D6NW	279	310	344		
C4NW	254	281	313		
BLOOST-calculation					
$\hat{P}$ (MW)	649	1321	2206	3537	4580
E(MW – sec)	19	26	32	40	45
$\hat{T}$ (°C) (D4NE)	558	701	820	954	1030
$\overline{T}$ core (°C)	201	252	300	356	387
$\hat{T}_{03}$ (°C)					
D6NW	260	313	358	405	436
C4NW	201	241	276	316	341

(a) Pulse data taken from test pulses performed on November 21, 2005

Table 20 Historical Pulsing Data for WSU Core 34A

Pulse number	Date	Reactivity added	$\hat{T}_{0.3}$ D6NW	$\hat{T}_{0.3}$ C4NW	Peak Power (MW)	Energy (MW•s)
1040	11/21/2005	1.25	242	227	60	11.5
1041	11/21/2005	2.00	332	317	1200	24
1043	11/21/2005	1.50	279	254	-----	16
1044	11/21/2005	1.50	279	254	240	16
1045	11/21/2005	1.75	310	281	440	20
1046	11/21/2005	2.00	344	313	1030	25
1047	12/5/2005	1.25	241	217	120	12.4
1048	5/31/2006	1.25	246	223	120	11.6
1049	5/31/2006	1.75	305	279	500	17.2
1050	5/31/2006	2.00	378	309	1000	20
1051	11/6/2006	1.25	259	230	160	12
1052	11/6/2006	1.50	292	252	260	13
1053	11/6/2006	1.75	319	286	480	17.2
1054	11/6/2006	2.00	355	317	----	21
1055	11/6/2006	2.15	375	334	1420	25
1056	11/6/2006	2.15	382	340	1420	24
1057	12/14/2006	1.75	317	281	1200	----
1058	1/29/2007	1.25	261	228	190	11.9
1059	1/29/2007	2.15	376	335	1420	24.8
1060	1/29/2007	2.00	355	315	1000	22
1061	1/29/2007	0.75	92	78	0	0
1062	1/29/2007	1.01	218	188	399	7.8
1063	1/29/2007	1.03	222	193	399	8.3
1064	5/14/2007	1.25	256	-----	160	11
1065	5/14/2007	2.00	354	316	1000	21.5
1066	5/14/2007	1.50	292	259	220	14.3
1067	5/14/2007	1.75	320	284	440	16.8
1068	5/14/2007	1.25	257	229	----	10.5
1069	5/14/2007	1.50	290	257	210	14.2
1070	8/7/2007	1.50	294	260	300	15

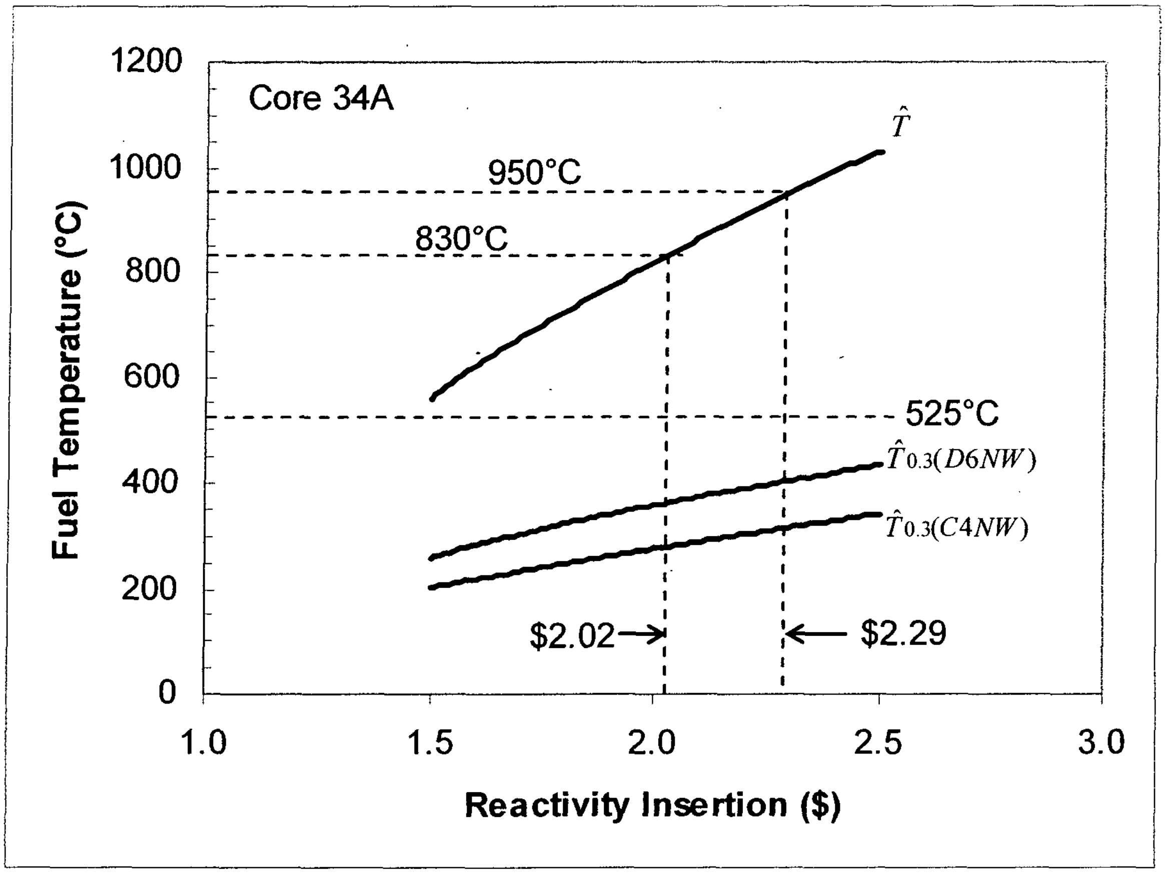

Figure 33 WSU Mixed HEU Core 34A – Pulse Performance at Current - Operating Conditions.

4.5.11 Pulse Operation – WSU Core Mixed LEU Core 35 A – BOL

The WSU reactor has had extensive experience with pulsing performance using fuel having a strong temperature dependent prompt negative temperature coefficient of reactivity (α). The new LEU (30/20) fuel also produces a similarly strong temperature dependence of α. Comparing the curves for α for FLIP fuel, Figure 28, and for LEU (30/20) fuel, Figure 29, one notes similar temperature dependences; however, the magnitude of α is somewhat smaller for the LEU fuel. The BOL neutron lifetime is 28.2 µsec; the EOL neutron lifetime is 27.0 µsec.

Table 21 presents the beginning of core life pulsing parameters for the WSU Mixed LEU Core 35A. Results for the WSU Mixed HEU Core 34A are included for easy comparison. Results for reactivity insertions of $1.50, $1.75, $ 2.00, $2.30, $2.50, $2.75 and $3.19 are shown. The peak fuel temperature in the core (D4NE) is listed. Calculated temperature results ($\hat{T}_{0.3}$) are shown for the instrumented fuel elements in positions D6NW and C4NW. Figure 34 shows a plot of the pulsed fuel temperatures as a function of reactivity insertion for the WSU Mixed LEU Core 35A fuel at BOL.

Since the peak pulsed fuel temperature will be limited to 830°C during a pulse, the reactivity insertion ($2.04) to produce this temperature is also indicated in Figure 34. Table 21 also shows that the maximum insertion of the transient-rod reactivity required to reach the fuel temperature Safety Limit of 1150°C is slightly larger than $2.75.

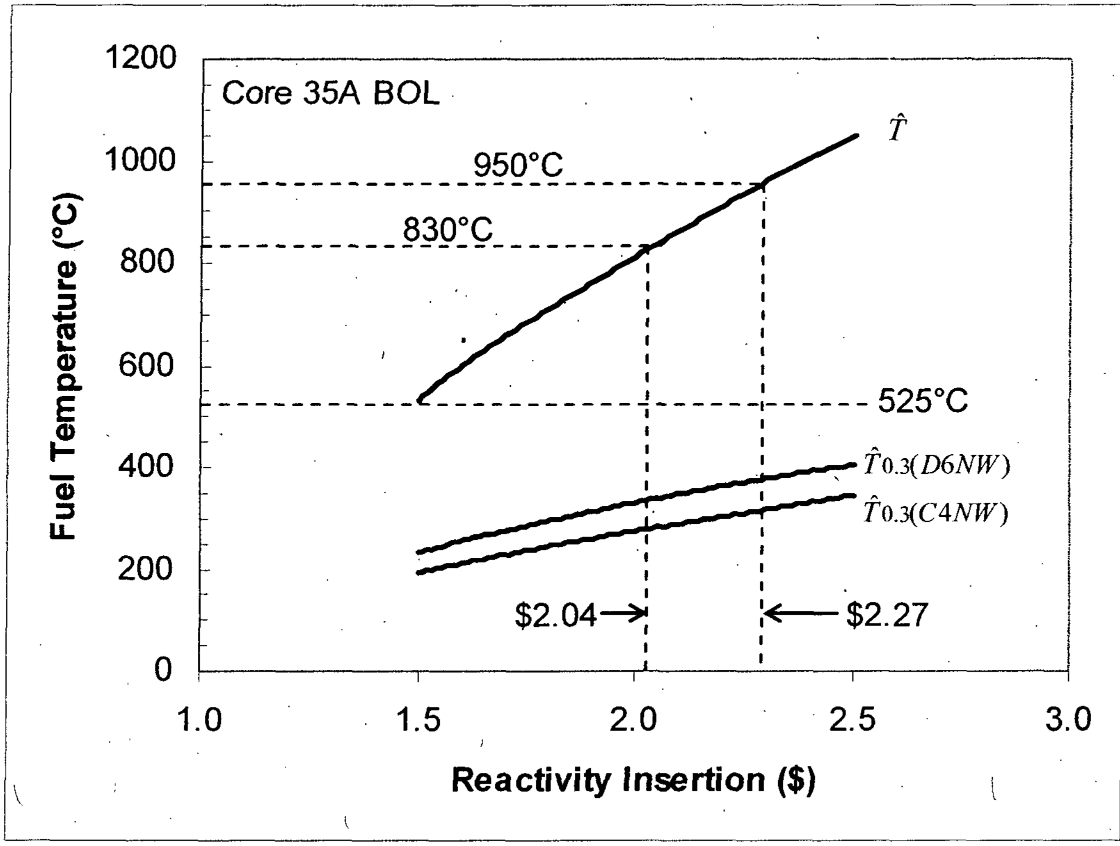

Figure 34 Pulse Performance at Beginning of Core Life, WSU Mixed LEU Core 35A

Table 21 Calculated Pulse Performance for WSU Mixed HEU Core 34A, Current Performance, and WSU Mixed Core 35A, BOL

Parameter	WSU Mixed HEU Core 34A – Current					WSU Mixed LEU Core 35A – BOL						
	\$1.50	\$1.75	\$2.00	\$2.30	\$2.50	\$1.50	\$1.75	\$2.00	\$2.30	\$2.50	\$2.75	\$3.19
$\hat{P}$ (MW)	649	1321	2206	3537	4580	546	1143	1969	3222	4210	5506	6686
E (MW-sec)	19	26	32	40	45	16	22	29	36	41	46	52
$\hat{T}$ (°C) (D4NE)	558	701	820	954	1030	533	683	812	961	1046	1132	1234
$\overline{T}_{core}$ (°C)	201	252	300	356	387	174	227	276	334	367	401	442
$\hat{T}_{0.3}$ (°C)												
D6NW	260	313	358	405	436	232	285	331	379	405	437	479
C4NW	201	241	276	316	341	192	236	275	318	345	375	403

Table 22 Calculated Pulse Performance for WSU Mixed LEU Core 35A, BOL, and WSU Mixed Core 35A, EOL

Parameter	WSU Mixed LEU Core 35A – BOL							WSU Mixed LEU Core 35A - EOL						
	\$1.50	\$1.75	\$2.00	\$2.30	\$2.50	\$2.75	\$3.19	\$1.50	\$1.75	\$2.00	\$2.30	\$2.50	\$3.00	\$3.19
$\hat{P}$ (MW)	546	1143	1969	3222	4210	5506	6686	506	1099	1914	3172	4202	6373	6816
E (MW-sec)	16	22	29	36	41	46	52	16	22	29	36	42	52	55
$\hat{T}$ (°C) (D4NE)	533	683	812	961	1046	1132	1234	465	608	733	878	961	1119	1154
$\overline{T}_{core}$ (°C)	174	227	276	334	367	401	442	168	224	275	337	373	441	455
$\hat{T}_{0.3}$ (°C)														
D6NW	232	285	331	379	405	437	479	206	256	301	350	379	430	444
C4NW	192	236	275	318	345	375	403	227	281	330	381	408	473	489

4.5.12 Pulse Operation – WSU Mixed LEU Core 35A – EOL

The BLOOST code has been used to calculate the pulsing performance of the WSU Mixed LEU Core 35A at EOL, ~1000 MWD burnup. The procedure is the same as used above for BOL conditions. Results are shown in Table 22 for reactivity insertions of \$1.50, \$1.75, \$2.00, \$2.30, and \$2.50, \$3.00 and \$3.19. Results are presented for peak pulsed power, integrated energy, peak fuel temperature in the hottest fuel rod, average reactor core temperature, and predicted thermocouple temperatures.

Figure 35 illustrates the dependence of fuel temperatures on reactivity insertions at 1000 MWD burnup. As can be seen from Figure 35, a reactivity insertion of \$2.20 is required at the EOL conditions of ~1000 MWD burnup to reach the limiting temperature condition of 830° C. In view of the fact that the peak fuel temperatures ($\hat{T}$) at 1000 MWD are lower rather than higher than the initial peak fuel temperature, the Limiting Reactivity Insertion for pulse operation can safely remain at \$2.04, as determined earlier.

Similarly, the maximum possible accidental pulse of \$3.19 (calculated worth at transient-rod) is required to just reach the Safety Limit of 1150°C. This is slightly higher than the \$2.75 insertion required at BOL conditions as indicated in Table 21.

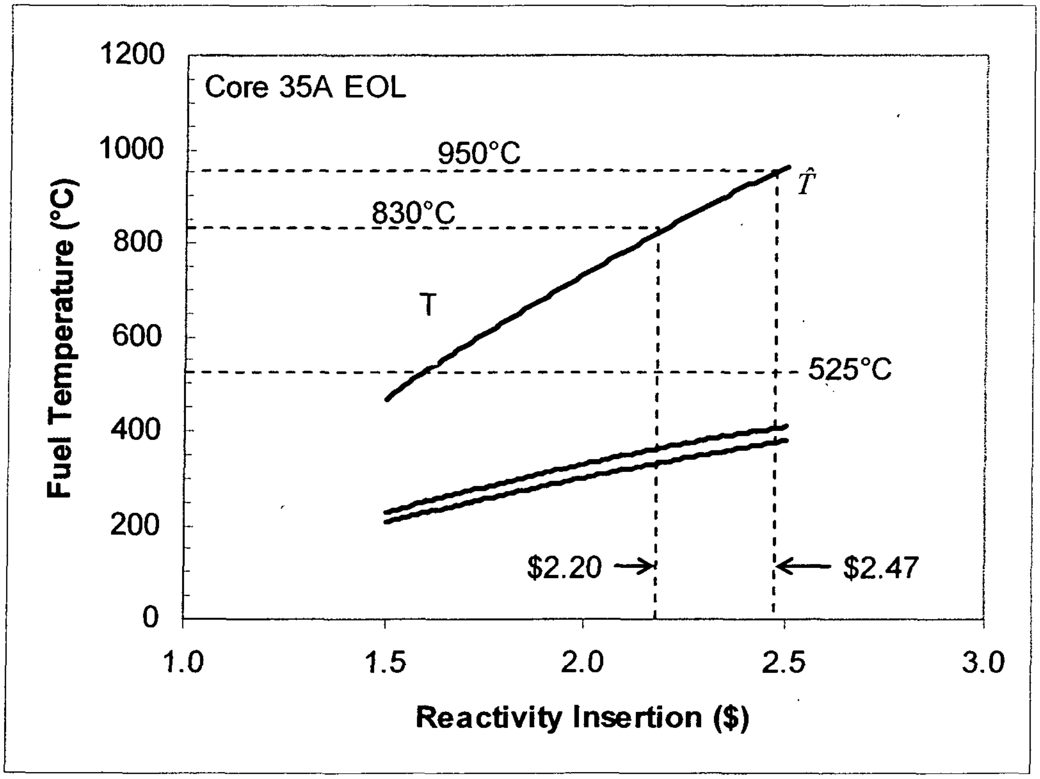

Figure 35 Pulse Performance at 1000 MWD Burnup – WSU Mixed LEU Core 35A

4.6 Functional Design of the Reactivity Control System

No changes in the reactivity control system are required.

4.7 Thermal-Hydraulic Analysis – WSU Core 34A

4.7.1 Analysis of Steady State Operation.

The following evaluation has been made for a TRIGA system operating with cooling from natural convection water flow around the fuel elements. In this study, the predicted steady state thermal-hydraulic performance of the WSU Mixed HEU Core 34A, at current operating conditions, was determined for the reactor operating at 1.0 MW and a water inlet temperature of 30°C. Operational data are used for the benchmark comparisons. An average powered fuel rod and a maximum powered fuel rod were analyzed. The RELAP5 computer code, (Reference 9), was used to determine the natural convection flow rate, the coolant and clad axial temperature profiles, and the clad wall heat flux axial profile. The reactor power at which critical heat flux is predicted to occur was calculated with the aid of the RELAP5 code. The TAC2D thermal analysis code, (Reference 10), was used to determine the fuel average and fuel maximum temperatures.

4.7.2 RELAP5 Code Analysis and Results.

RELAP5 is a computer program for calculating thermal hydraulics in nuclear reactors. In this application it is used to calculate the natural convection flow through a vertical water coolant channel bounded by cylindrical heat sources. Output from the RELAP5 code includes: channel flow rate, outlet velocity, temperature rise of the fluid along the channel, maximum heat flux and maximum clad temperature. The assumption is made that there is no cross flow between adjacent channels. Input to the program includes the following:

1) Size and spacing of the heat sources;

2) Axial heat source distribution;

3) Pool height above the core;

4) Inlet and exit pressure loss coefficients;

5) Inlet water temperature.

Analysis is performed on two flow channels divided into axial segments that represent a single fuel element and the entire core. The natural convection system for the WSU TRIGA reactor was based on the 4-rod cluster of fuel elements. The representation used herein establishes one flow channel bounded by four fuel elements. The reactor geometry, power factors and hydraulic data for the RELAP5 input are given in Table 23.

Table 23 RELAP5 Input for Reactor and Core Geometry and Heat Transfer, WSU

Core and Reactor Geometry	
Unheated core length at inlet, mm	100
Unheated core length at outlet, mm	124
Distance from top of pool surface to top of core, mm	6540
Hydraulic Data	
Inlet pressure loss coefficient	2.02
Exit pressure loss coefficient	1.38
Ambient pressure at pool surface, MPa	0.101

A RELAP5 thermal hydraulic analysis was done for an average flow channel, a maximum powered channel and the two IFE channels. The analysis was conducted by considering the hydraulic characteristics of a typical flow channel represented by the geometric data given in Table 24. The thermal and hydraulic parameters for the WSU Mixed HEU Core 34A are given in Table 25.

Table 24 Hydraulic Flow Parameters For a Flow Channel

Flow area (mm^2/rod)	501.5
Wetted perimeter (mm/rod)	112.6
Hydraulic diameter (mm)	17.82
Fuel element heated length (mm)	381
Fuel element diameter (mm)	35.842
Fuel element surface area (mm^2)	4.29×10^4

The heat generation in the fuel element is distributed axially in a piece-wise fashion to represent the curves in Figure 31. There are 119 fuel elements in the initial core, 51 FLIP HEU SFEs and 68 partially burned 8.5/20 LEU SFEs. The hot-rod power ratio is 2.490, Table 16, for hot core conditions.

The driving force is supplied by the buoyancy of the heated water in the core. Countering this force are the contraction and expansion losses at the entrance and exits to the channel, and the acceleration and potential energy losses and friction losses in the cooling channel itself. The pressure drops through the flow channel are dependent on the flow rate while the available static driving pressure is fixed for a known core height and ambient pressure.

A summary of the RELAP5 results for WSU is given in Table 26.

Table 25 TRIGA Thermal and Hydraulic Parameters for WSU Mixed HEU Core 34A, 1.0 MW

Parameter	Initial Core
Number of fuel elements	119
Diameter, mm (in.)	35.842 (1.411)
Length (heated), mm (in.)	381 (15.0)
Core flow area, mm^2 (ft^2)	59677 (0.64236)
Core wetted perimeter, mm (ft.)	13,399 (43.96)
Flow channel hydraulic diameter, mm (ft.)	17.82 (0.05845)
Core heat transfer surface, m^2 (ft^2)	5.105 (54.95)
Hot rod factor	2.490
Axial peaking factor	1.290
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	114 (238)
Exit coolant temperature (average), °C (°F)	61.21 (142.2)
Exit coolant temperature (maximum), °C (°F)	84.31 (183.8)
Coolant mass flow, kg/sec (lb/hr)	7.67 (60,836)
Average flow velocity, mm/sec (ft/sec)	130 (0.425)
Peak fuel temperature in average fuel element, °C (°F)	296 (564)
Maximum wall temperature in hottest element, °C (°F)	142.7 (288.9)
Peak fuel temperature in hottest fuel element, °C (°F)	435 (815)
Core average fuel temperature, °C (°F)	221 (429)
Average heat flux, W/cm^2 ($BTU/hr\text{-}ft^2$)	19.59 (62,112)
Maximum heat flux in hottest element, W/cm^2 ($BTU/hr\text{-}ft^2$)	62.75 (198,954)
Minimum DNB ratio	2.47

Table 26 Steady State Results for WSU Core 34A, 1.0 MW

Initial Core (119 elements – 51 FLIP HEU SFEs, 68 Burned 8.5/20 LEU SFEs)	**Average Rod**	**Maximum Rod**
Channel natural convection mass flow rate, kg/sec	0.0644	0.0921
Exit coolant flow temperature, °C	61.21	84.31
Maximum wall heat flux, W/cm^2	25.92	62.75
Maximum flow velocity, cm/sec	13.07	18.95
Maximum clad temperature, °C	130.1	142.7
Exit clad temperature, °C	121.1	130.6

The RELAP5 code also calculates the critical heat flux, i.e. the heat flux at which there is a departure from nucleate boiling (DNB) and the transition to film boiling begins. The correlation used in RELAP5 Mod. 3.2 to calculate this heat flux is the Groeneveld 1986 Correlation. (Reference 11) A second correlation historically used by TRIGA reactors is due to Bernath. (Reference 12) Bernath gives a lower value for the critical heat flux compared to the 1986 Groeneveld correlation used in RELAP5 Mod. 3.2. The lower critical heat flux from the Bernath correlation is used here for determining the minimum DNB ratio, i.e. the minimum ratio of the local allowable heat flux to the actual heat flux.

The RELAP5 code analysis has been run for the critical heat flux for the WSU Core 34A reactor operating at 1.0 MW at benchmark conditions. The data was obtained using an inlet temperature of 30°C and systematically increasing the reactor power until RELAP5 indicated a DNB ratio equal to one based on the Bernath correlation. The maximum power per fuel element for which the DNB ratio is 1, is 51.7 kW/element. For a core with a rod peaking factor of 2.490, this maximum fuel element power corresponds to a maximum reactor power of 2.47 MW (51.7 kW per rod/2.490 × 119 rods = 2.47 MW). Hence, the DNB ratio for the WSU Core 34A at the stated conditions is 2.47. The minimum DNB ratio is 1.9 at a power level of 1.3 MW.

4.7.3 TAC2D Fuel Temperature Analysis and Results.

The TAC2D general purpose heat conduction code was used to calculate steady state maximum and average fuel temperatures for the average powered rod, the maximum powered rod, and the two IFEs. A radial-axial (R,Z) two-dimensional model of the center zirconium rod (6.35 mm diameter), the fuel annulus, the fuel-to-clad gap, and the 0.5 mm thick stainless steel cladding of a single fuel pin was constructed. The model included only the active length of the fuel pin.

TAC2D is a code for calculating steady-state and transient temperatures in two-dimensional problems by the finite difference method. The configuration of the body to be analyzed is described in the rectangular, cylindrical, or circular coordinate system by orthogonal lines of constant coordinate called grid lines. The grid lines specify an array of nodal elements. Nodal points are defined as lying midway between the bounding grid lines of these elements. A finite difference equation is formulated for each nodal point in terms of its heat capacity, heat generation and heat flow paths to neighboring nodal points.

The TAC2D code requires as input a geometric description of the problem and properties of the materials considered. The radial and axial power distributions in the fuel are also provided as input. The problem is defined in cylindrical R-Z geometry. The axial distribution of the surface heat transfer coefficient and coolant temperature from the RELAP5 code is used to model the outer radial boundary. The fuel-to-clad interface conductance assumes the fuel pin is sealed with air. In order to account for gap closure due to fuel swelling, the cold gap was assumed to be 0.2 mils throughout the core. Based on the comparison of calculated versus measured reactivity loss in Section 4.5.7, it appears that the average core gap is slightly larger than 0.2 mils. Some additional gap closure occurs due to the relative expansion of the fuel and cladding at normal operating temperatures.

A summary of the TAC2D results for WSU Core 34A has been given in Table 27 for 0.5, 1.0, and 1.3 MW operations with 119 fuel elements.

Table 27 Calculated and Measured Fuel Temperatures, WSU Mixed HEU Core 34A

P(MW)	$\hat{T}_{meas}$ (°C)	T_{calc} (°C)			
		$\hat{T}$	$\hat{T}_{0.3}$		$\overline{T}_{avg\ core}$
			D6NW	C4NW	
0.5		332	255	258	156
1.0	302-304	435	318	321	221
1.3		520	371	375	234

4.8 Thermal Hydraulic Analysis – WSU Mixed LEU Core 35A

4.8.1 Analysis of Steady State Operation – WSU Mixed LEU Core 35A.

The following evaluation has been made for the TRIGA fuel system with 4-rod configuration operating with cooling from natural convection water flow through 4-rod clusters of fuel. The steady state thermal-hydraulic performance of the WSU Core 35A was determined for operation at 1.0 MW with a water inlet temperature of 30°C.

An average powered fuel rod and a maximum powered fuel rod were analyzed. The RELAP5 computer code was used to determine the natural convection flow rate, the coolant and clad axial temperature profiles, and the clad wall heat flux axial profile. The RELAP5 code was also used to determine the clad wall maximum heat flux versus coolant inlet temperature for departure from nucleate boiling. The TAC2D thermal analysis code was used to determine the fuel average and fuel maximum temperatures.

4.8.2 RELAP5 Code Analysis and Results – WSU Mixed LEU Core 35A

The RELAP5 analysis was performed using the method outlined in Section 4.7. The reactor geometry and hydraulic data for the RELAP5 input are given in Table 23. The natural convection system for the WSU TRIGA was based on the 4-rod cluster of fuel elements. The representation used herein establishes one flow channel bounded by four fuel elements.

A RELAP5 thermal hydraulic analysis was done for an average flow channel and a maximum powered channel. The analysis was conducted by considering the hydraulic characteristics and flow parameters of a typical flow channel represented by the geometric data given in Table 24. The thermal and hydraulic parameters for the WSU Mixed LEU Core 35A are given in Table 28.

The heat generation in the fuel element is distributed axially in a piece-wise fashion to represent the curves in Figure 32. It is further given that there are 119 fuel elements in the initial core, 51 – 30/20 LEU SFEs AND 68 partially burned 8.5/20 SFEs. The hot-rod power ratio is assumed to be 2.474 for hot core conditions.

A summary of the RELAP5 results for the 1.0 MW WSU Core 35A is given in Table 29.

Table 28 TRIGA Thermal and Hydraulic Parameters for WSU Mixed LEU Core 35A, 1.0 MW

Parameter	**Initial Core**
Number of fuel elements	119
Diameter, mm (in.)	35.842 (1.411)
Length (heated), mm (in.)	381 (15.0)
Core flow area, mm^2 (ft^2)	59,677 (0.64236)
Core wetted perimeter, mm (ft.)	13,399 (43.96)
Flow channel hydraulic diameter, mm (ft.)	17.82 (0.05845)
Core heat transfer surface, m^2 (ft^2)	5.105 (54.95)
Hot rod factor	2.474
Axial peaking factor	1.286
Inlet coolant temperature, °C (°F)	30 (86)
Coolant saturation temperature, °C (°F)	114 (238)
Exit coolant temperature (average), °C (°F)	61.22 (142.2)
Exit coolant temperature (maximum), °C (°F)	84.06 (183.3)
Coolant mass flow, kg/sec (lb/hr)	7.67 (60,858)
Average flow velocity, mm/sec (ft/sec)	130 (0.425)
Peak fuel temperature in average fuel element, °C (°F)	399 (751)
Maximum wall temperature in hottest element, °C (°F)	142.6 (288.6)
Peak fuel temperature in hottest fuel element, °C (°F)	500 (932)
Core average fuel temperature, °C (°F)	303.7 (578.7)
Average heat flux, W/cm^2 (BTU/hr-ft^2)	19.59 (62,112)
Maximum heat flux in hottest element, W/cm^2 (BTU/hr-ft^2)	62.12 (196,970)
Minimum DNB ratio	2.45

Table 29 Steady State Results for WSU Core 35A, 1.0 MW

Initial Core (119 elements)	**Average Rod**	**Maximum Rod**
Channel natural convection mass flow rate, kg/sec	0.0644	0.0919
Exit coolant flow temperature, °C	61.22	84.06
Maximum wall heat flux, W/cm^2	25.93	62.12
Maximum flow velocity, cm/sec	13.08	18.90
Maximum clad temperature, °C	129.7	142.6
Exit clad temperature, °C	121.6	130.9

4.8.3 TAC2D Fuel Temperature Analysis and Results – WSU Mixed LEU Core 35A

Using the methods given in Section 4.7.3, a TAC2D analysis was performed for the 1.0 MW WSU Core 35A for operation with 119 fuel elements.

The fuel temperatures for WSU Mixed LEU Core 35A steady state operation have been calculated for the hottest, measured, and average fuel rods. These results are presented in Table 30. The value for $\hat{T}_{0.3}$ is the thermocouple temperature that is located 0.3 inch from the fuel centerline.

Table 30 Calculated Fuel Temperatures, WSU Mixed LEU Core 35A

P (MW)	$\hat{T}$ (°C)	$\hat{T}_{0.3}$ (°C)		$\overline{T}_{core}$ (°C)
		D6NW	**C4NW**	
0.5	407	350	361	242
1.0	500	427	440	304
1.3	541	457	469	327

It will be noted that neither of the instrumented fuel elements are located at the hottest core position. The IFEs are located at (D6NW) and (C4NW); the hottest fuel element is calculated to be (D4NE), Figure 27 which is adjacent to the transient rod. The guide tube for the transient rod has an OD of 1.485 in., which is slightly larger than the fuel element OD of 1.411 in. and as a result reduces the flow area of the maximum powered fuel rod by about 3.5%. In calculations done for the Texas A&M 1MW HEU to LEU conversion this reduction in flow area resulted in about a 3.3% decrease in the MDNBR; similar results are expected for WSU. The sensing tip of the thermocouple is 0.3 in. (7.62 mm) from the fuel axial center line, just outside the 0.25 in. (6.35 mm) diameter zirconium rod positioned along the axial center of the fuel. The results reported in Table 30 give the peak fuel temperature $\hat{T}$ in the hottest fuel element, the computed temperature $\hat{T}_{0.3}$ in the IFEs that can be compared with future measured temperatures, and the average core temperature $\overline{T}_{core}$.

The manufactured radial gap between the fuel and cladding is limited to a maximum of 2 mils and the average radial gap is limited to less than 1.75 mils. The TAC2D analysis for the maximum and IFE temperatures is based on the maximum cold gap of 2 mils and the average temperature is based on the maximum average gap of 1.75 mils. As the TRIGA core operates, offset swelling of the fuel tends to close the gap as was observed in the conversion of the Texas A&M TRIGA reactor. The effect of a closing gap on fuel temperatures is presented in Figure 36.

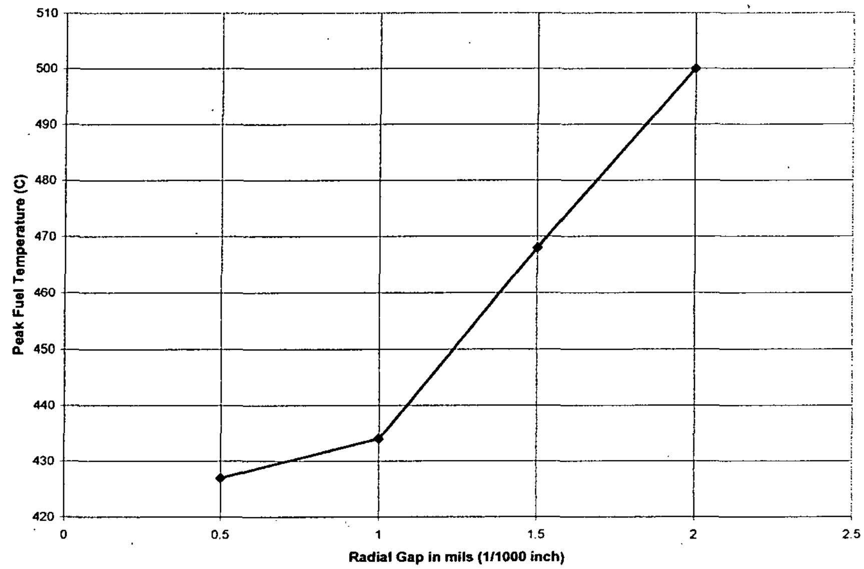

Figure 36 Peak Fuel Temperature as a function of Manufactured Radial Gap

The average power per fuel element in the WSU Mixed LEU Core 35A operating at 1000 kW is 8.4 kW/element. The highest powered fuel element is located immediately adjacent to the transient rod in location D4NE and produces a power of 20.8 kW/element. An ideal location for one of the Instrumental Fuel Elements (IFE) would be this location. The current configuration for the WSU Mixed LEU Core 35A has retained the IFEs in the same location as for WSU Mixed HEU Core 34A. These are core location D6NW with a power of 10.7 kW/element, and core location C4NW with a power of 12.1 kW/element.

4.8.4 Steady-State Analysis Results Summary – WSU Mixed LEU Core 35A

Table 28 lists the pertinent heat transfer and hydraulic parameters for the 1.0 MW WSU Mixed LEU Core 35A. Results are presented therein for an average channel and a maximum powered channel (hot channel) at initial core conditions. Also shown are the peak fuel temperatures in the hottest fuel element and average fuel element calculated with the TAC2D code.

The RELAP5 code analysis has been run for the critical heat flux for the WSU Mixed LEU Core 35A operating at 1.0 MW at BOL conditions. The data was obtained by using an inlet temperature of 30°C and then systematically increasing the reactor power until RELAP5 indicated a DNB ratio equal to one based on the Bernath Correlation. The maximum power per fuel element for which the DNB ratio is 1, is 52 kW/element. For a core with a rod peaking factor of 2.474, this maximum fuel element power corresponds to a maximum reactor power of 2.45 MW. Hence, the DNB ratio for the WSU Core 35A at the stated conditions is 2.45. The minimum DNB ratio is 1.88 at a power level of 1.3MW.

4.9 Thermal Neutron Flux Values, WSU Mixed LEU Core 35A

4.9.1 Thermal Neutron Flux Values in LEU Core

The DIF3D code provides neutron flux values. Figure 37 presents a 3-D representation of the thermal neutron distribution throughout the core and into the surrounding water and graphite.

Figure 38 shows the flux plot through the transient rod in a direction perpendicular to the face of the thermal column/BNCT Filter box. In the region between 5 cm and 35 cm the flux through the partially burned LEU 8.5/20 fuel with the peak in the water of the control blades is seen. In the region between 35 cm and 135 cm the flux through the fresh 30/20 LEU fuel (flux depressed) is seen with the water peak in the transient rod. Finally in the region between 135 cm and 155 cm the flux through the opposite region of the partially-burned 8.5/20 LEU fuel with the peak in the area of the control blades is seen.

Figure 39 presents a graphical representation of the neutron flux across the core through the transient rod in a direction parallel to the face of the thermal column/BNCT Filter box. This plot starts in the water reflector/shield, crosses the reactor, and ends in the water reflector on the other side of the reactor core. In this orientation the flux depression in the fresh 30/20 LEU fuel is clearly evident in the central region of the plot with the peaking in the water hole for the transient rod. The depression in the fresh 30/20 LEU fuel is caused by the erbium and large uranium loading as it was evident in the Texas A&M post conversion flux measurements.

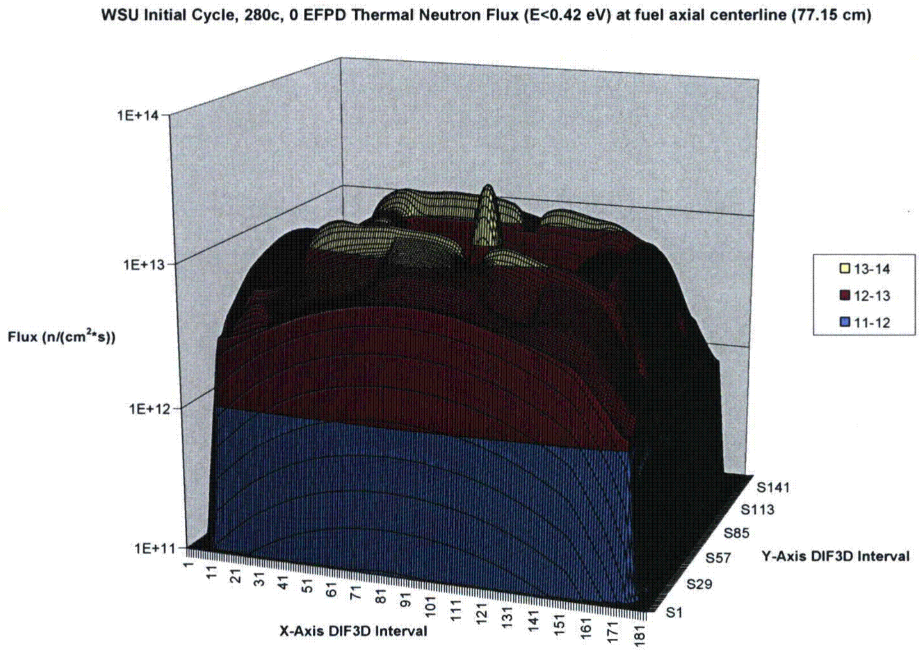

Figure 37 Thermal Neutron Flux (E<0.42 eV) at fuel axial centerline, BOL, WSU Mixed LEU Core 35A

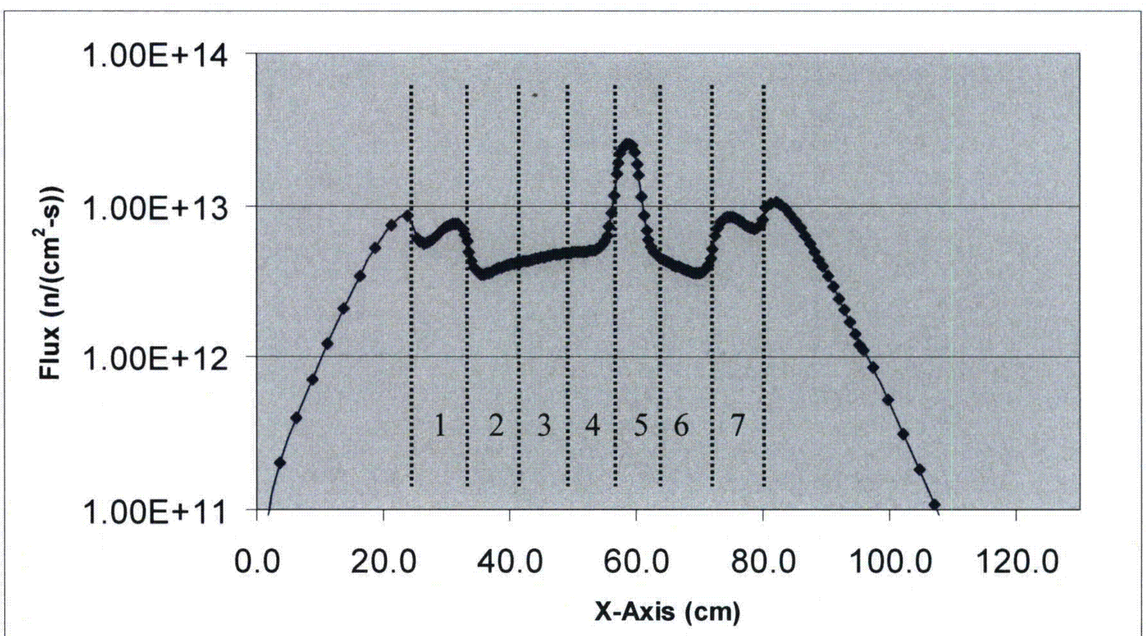

Figure 38 Thermal Neutron Flux (E<0.42 eV) Across the Core, West to East, through the Transient Rod at fuel centerline, BOL, WSU Mixed LEU Core 35A

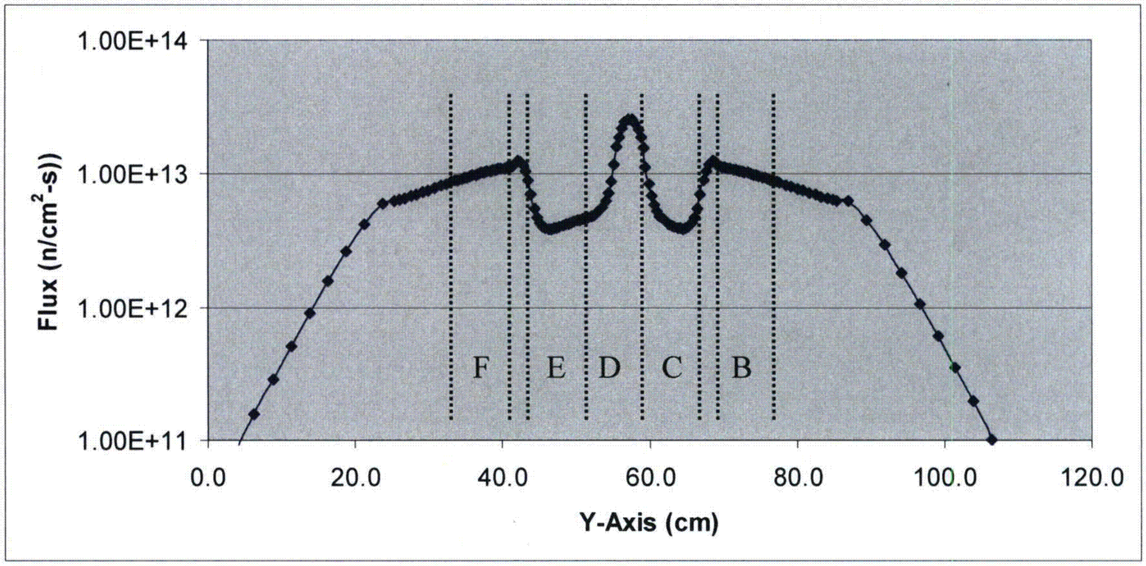

Figure 39 Thermal Neutron Flux (E<0.42 eV) Across the Core, South to North, through the Transient Rod at fuel axial centerline, BOL, WSU Mixed LEU Core 35A

5.0 Reactor Coolant System

The HEU to LEU conversion does not require any changes to the reactor coolant system.

6.0 Engineering Safety Features

The HEU to LEU conversion does not require any changes to the engineered safety features.

7.0 Instrumentation and Control

The HEU to LEU conversion does not require any changes to the instrumentation and control system.

8.0 Electrical Power System

The HEU to LEU conversion does not require any changes to the electrical power systems.

9.0 Auxiliary System

The existing procedure will be used for fuel storage. Criticality aspects of the existing storage arrangements have been verified and are included in Appendix B of this document.

10.0 Experimental Facility and Utilization

The HEU to LEU conversion does not require any changes to experimental facilities.

11.0 Radiation Protection and Radioactive Waste Management

The HEU to LEU conversion does not require any changes to the radiation protection and radioactive waste management.

12.0 Conduct of Operation

To be supplied by licensee.

12.1 Organization and Staff Qualification

The HEU to LEU conversion does not require any changes to the organization and staff qualification of the WSU Reactor Personnel

12.2 Procedures

The HEU to LEU conversion does not require any fundamental changes to the WSU reactor standard operating procedures with the exception of any references to HEU/FLIP or standard fuel

12.3 Operator Training and Re-qualification

The HEU to LEU conversion does not require any changes to the WSU reactor operator training and re-qualification program with the exception of updating training materials to describe new fuel type and accident analysis.

12.4 Emergency Plan

The HEU to LEU conversion does not require any changes to the existing WSU reactor Emergency Plan.

12.5 Physical Security

The HEU to LEU conversion does not necessitate any changes to the existing security plan at the time of conversion. However, WSU anticipates that there will be a need to make changes, for a period of time, when the HEU is removed from the core, stored, and until it is suitable for shipping. It is anticipated that changes to the plan will need to be made to comply with 10CFR73.60 and 10CFR73.67(d) during the time the fuel is no longer self-protected until the time that it is ultimately shipped. These proposed changes will be submitted by WSU under separate cover and withheld from public disclosure.

12.6 Reactor Reload Consideration

The WSU Mixed LEU Core 35A will have the capability to operate at 1.0 MW on demand for about 13 years following a weekly schedule of ≤35 MWHr. During the life of Core 35A it is conceivable that at some fuel may be introduced in the partially burned core. The following are most likely scenarios for the introduction of the new fuel:

1. Some of the partially burned 8.5/20 fuel becomes fully burned and must be replaced
2. A fresh IFE is introduced because the thermocouples in one of the in-core IFEs have failed
3. A fresh 30/20 LEU SFEs is introduced

Case 1 above is straightforward because the power factors in the region of the core where the partially burned 8.5/20 fuel is located are very low. Therefore introducing another partially burned 8.5/20 SFE from the irradiated fuel already on hand at WSU into the same core region will not lead to peaking factors that exceed values considered in this analysis.

Cases 2 and 3 on the other hand could lead to possible higher peaking factor especially if the fresh 30/20 IFE/SFE is located near the transient rod water hole. This will be most likely during pulsing operation. Therefore, depending on the configuration of the core at the time that the fresh fuel is added, core locations 6C, 6E, or 2D should be considered for introducing any new fuel. These locations have very low peaking factors both at the beginning and the end of life.

12.7 LEU Startup Plan

A detailed Startup Plan together with Acceptance Criteria is presented in Appendix A.1.

13.0 Safety and Accident Analysis

13.1 Safety Analysis

13.1.1 General Discussion and Summary

The safety of TRIGA fuel is due entirely to its design features. The safety features of a standard TRIGA fueled core are well known. Each of the 30/20 LEU fuel elements is designed to replace a FLIP HEU (8.5 wt % U, 70% enriched) element, with regards to reactivity; that is, 100 fresh 30/20 LEU fuel elements in a compact configuration is intended to have about the same core excess reactivity as 100 of the more heavily loaded FLIP TRIGA fuel elements.

As part of the Reduced Enrichment for Research and Test Reactors (RERTR) program, various tests were performed on high-uranium content, low-enriched TRIGA fuels and the test results submitted to the NRC. The NRC concluded in their Safety Evaluation Report, (Reference 2) that both the 20/20 and 30/20 uranium-zirconium hydride fuels "are generally acceptable for use in other licensed TRIGA reactors, with the provision that case-by-case analyses discuss individual reactor operating conditions in applications for authorization to use them".

In the present document, it is shown that one-for-one fresh TRIGA FLIP fuel and fresh TRIGA LEU (30/20) fuel behave very similar as regards cold, clean critical. However, both these types of TRIGA fuel react strongly to in core water filled regions. For the WSU analysis the transient rod is water followed and was properly modeled both for the steady state and pulsing operations.

13.2 Safety Limits

The safety of the operating TRIGA reactor system with LEU (30/20) fuel is related directly to the maximum temperature of the fuel and the continued availability of coolant. As demonstrated for all TRIGA fuel elements, the Safety Limit for water cooled fuel is taken conservatively as 1150°C. The Safety Limit for these fuel elements when air cooled is 950°C.

As analyzed in this report, all proposed reactor operations will involve low fuel temperatures with large margins of safety. The peak fuel temperature at steady state operation at 1.0 MW is 500°C. In normal pulsing, 830°C has been chosen as the limit for the peak fuel temperature and administrative controls are in place to maintain the limit on prompt reactivity insertions. The limiting peak pulsing fuel temperature (830°C) has been chosen to address the problem of hydrogen migration resulting from long term high power operation. Both of these temperatures have large margins of safety to 1150°C.

The two power level scrams (125% of 1.0 MW) are used to assure that reactor power is limited to a level that yields acceptable fuel temperatures, as noted above.

The fuel temperature scram (1) (500°C) is maintained in all modes of operation. The high power level scrams (2) (125%) are effective in steady state mode of operation.

In addition to the protection provided by the several, redundant scrams, administrative procedures and written operating procedures contribute additional limits on operation to protect the reactor, the facility, and the public.

13.3 Evaluation of LSSS for WSU LEU (30/20) Fuel

13.3.1 Steady State Mode

The value of the Limiting Safety System Setting (LSSS) is chosen to prevent the TRIGA Safety Limit (1150°C) from being reached in any mode of operation. The limiting safety system setting in an instrumented fuel element has been selected as 500 °C. The location of the fuel cluster containing the instrumented fuel element shall be chosen to be as close as possible to the hottest fuel element in the core. (The hottest element in the core is (D4NE), Figure 4.2. In the present analysis, the instrumented fuel elements (IFE) are located in core location D6NW and C4NW. (Other locations can be chosen for the IFE.) The LSSS temperature setting is smaller than the Safety Limit temperature to account for several factors, including:

i. Accuracy of temperature calibration
ii. Precision of electronic readout/scram circuitry
iii. Account taken of location of sensing tip of thermocouple 0.3 inch from axial center line of IFE
iv. Difference in peak temperature in IFE compared to that in the hottest fuel element

Safety Margin

The basis for selecting 500°C as the limiting safety system temperature setting is the following. The limiting safety system setting is a temperature which, if exceeded, shall cause a reactor scram to be initiated, preventing the safety limit from being exceeded. A peak core temperature of 950°C in LEU fuel is the criteria established to provide a minimum safety margin of 200°C for all modes of operation. A part of this margin is used to account for the difference between the maximum core temperature and measured temperatures resulting from the actual location of the thermocouple, 0.3 inches from the axial center line of the fuel element. If the instrumented fuel element were located in the hottest position in the core, the difference between the true and measured temperatures would be small. However, as explained above, the IFEs have been maintained in core locations D6NW and C4NW. Calculations indicate that, for this case, the true temperature at the hottest location in the core at 1.0 MW will differ from the measured temperature by about 17.1% (500/427 = 1.1709) for the IFE in core location D6NW and 13.6% (500/440 = 1.1363) for the IFE in core location C4NW, (See Table 30). Thus, for the steady state mode of operation, if the temperature in the thermocouple element were to reach the trip setting of 500°C, the true temperature at the hottest location in the core would be less than 600°C, providing a safety margin of at least 550°C for LEU (30/20) type elements. At a steady state reactor power of 1.3 MW peak fuel temperature $\hat{T}$ in the hottest fuel in the WSU LEU (30/20) core would be at most 520°C. These resulting safety margins are ample to account for any remaining uncertainty in the accuracy of the fuel temperature measurements and any overshoot in reactor power resulting from a reactor transient during steady state mode operation.

13.3.2 Pulse Mode

In the pulse mode of operation, the same limiting safety system setting will apply. However, the temperature channel will have no effect on limiting peak powers generated because of the relatively long time constant (seconds) for the recorded temperature as compared with the width of the pulse (few milliseconds). In this mode, however, the temperature trip, if activated, will cause all scrammable rods to fall and will act to reduce the amount of energy generated in the entire pulse transient by cutting the "tail" of the energy transient even if the pulse rod remains stuck in the fully withdrawn position.

13.4 Maximum Allowable Pulsed Reactivity Insertion

In Sections 4.5.11 and 4.5.12, the pulsing performance of the LEU (30/20) core has been reviewed for both beginning of core life and at 2000 MWD burnup. At a core burnup of 1000 MWD, a reactivity insertion of \$2.04 will produce a peak fuel temperature; namely, ~830°C.

In view of the small change in peak fuel temperatures with core burnup, a maximum allowable pulsed reactivity insertion of \$2.00 is a reasonable choice.

13.5 Accident Analysis

13.5.1 Analysis Changes to DBA/MHA Event

The Maximum Hypothetical Accident (MHA) for a TRIGA reactor is defined as the loss of the integrity of the fuel cladding of one fuel rod in air. This MHA definition is consistent with the Design Basis Accident (DBA) endorsed by the NRC. NUREG/CR-2387, (Reference 13), suggests, and NRC accepts, that for a 1.0 MW TRIGA reactor, the DBA is the release in air of fission products from a single irradiated fuel element. The evaluated dose resulting from the fission product release from a single FLIP fuel element was found acceptable by the NRC. The loss of pool water is typically treated separately as a Loss of Coolant Accident (LOCA). The MHA for the WSU TRIGA reactor was originally analyzed in the Safety Analysis for converting the WSU TRIGA reactor to FLIP fuel of May 1974. (References 14 and 15) This analysis was revised in the 2002 SAR that was submitted to support WSUs application to renew its operating license. The revised analysis used the same basic data as used in the previous analysis but the radiological effects are calculated using more recent analysis methods and guidelines published by the Federal Government. (Reference 16)

In this section, the radiological impact of the loss of fission products from a single TRIGA fuel element is reviewed. To compare the relative abundance of fission products, the pertinent operating parameters are compared for a FLIP core and a LEU (30/20) core. The major parameter important to radiological impact is the fission product inventory due to burnup and power density. The WSU mixed HEU Core 34A has a life of about 1000

MWD and is essentially the same as the WSU mixed LEU Core 35A as discussed in Section 4.5.6. The energy burn up capability based on >50% of the U-235 has been evaluated as 77 MWD/fuel element for FLIP fuel. (Reference 13) For WSU LEU (30/20) fuel, the energy burnup capability based on >50% of the U-235, Ref 13 gives 57 MWD/fuel element.

The hottest 30/20 fuel element in the LEU core at the end of life will contain less activity than a FLIP element of similar burn up. Prior WSU SARs based the fission product inventory of the hottest fuel element using a power density of 30 kW per fuel rod and an infinite irradiation time. The 30 kW per fuel rod is much greater than the maximum predicted power density of 20.8 kW for WSU Mixed LEU Core 35A. The WSU SAR inventory was derived from the basic data of Pecking and King, (Reference 17), along with the documented fact that only gaseous fission products escape when the cladding of a TRIGA fuel rod ruptures. Table 31 compares the WSU SAR inventory with an ORIGEN calculation for 30/20 fuel using similar assumptions. In performing the ORIGEN calculations, maximum inventories occurred after 200 MWD of burn up. This burn up maximized buildup while minimizing the effect of fissile material depletion. Thus, the radiological impact from an MHA event on the members of the public and the facility workers will be essentially the same as that evaluated for FLIP fuel.

The WSU 2002 SAR submittal concluded that the consequences of the MHA show that the only significant worst case radiation exposure is the thyroid dose to a person in the pool room. The conditions necessary to produce this exposure are the failure of the cladding of one fuel rod along with a complete loss of pool water. The maximum possible radiation exposure to an individual outside the facility under the postulated conditions is minimal. The exposures are significantly below the generally acceptable accident results for non-power reactors of not more than 5 rem whole body and 30 rem thyroid for occupational exposure and not more than 0.5 rem whole body and 3 rem thyroid for members of the general public. In addition, the calculated accident exposures are well below the maximum values established in 20.1201 for occupational exposure and 20.1301 for public exposure. Thus, no realistic hazard to the staff at the reactor as well as the general public would result from the MHA. These conclusions are equally applicable to the WSU Mixed LEU Core 35A due to the similarity in fission product inventories.

A stand-alone MHA analysis was prepared for a response to a Request for Additional Information. The MHA analysis is included in this SAR as Appendix MHA: Analysis of Maximum Hypothetical Accident.

Table 31 Gaseous Fission Products in a Single TRIGA Fuel Rod

ISOTOPE	Saturated Inventory (Ci)	
	SAR 2002	GA 2006
Br-82	40	1
83	137	135
84	253	263
85	330	309
87	780	504
Total Br	1540	1211
I-130m	260	4
131	734	734
132	1115	1104
133	1672	1704
134	2027	1961
135	1546	1596
136	785	818
Total I	8139	7919
Kr-83m	137	135
85m	330	309
85	67	20
87	634	631
88	912	894
89	1115	1157
Total Kr	3195	3145
Xe-131m	7	7
133m	40	49
133	1672	1661
135m	457	296
135	1621	1136
137	1545	1554
138	1166	1607
Total Xe	6508	6309

13.5.2 Analysis of Changes to LOCA Event

A LOCA analysis was performed as part of the 1979 Safety Analysis for Conversion to FLIP Fuel Conversion. (Reference 1) The analysis concluded that TRIGA fuel rods operating with power densities up to 22.3 kW/rod for standard fuel and 23.5 kW/rod for FLIP fuel would not fail in the event of a loss of coolant accident.

The conditions for this analysis included:

1. Operation of the reactor for essentially an infinite length of time
2. Sudden and complete loss of coolant (pool water).
3. The loss of water will shut down the reactor, however, the decay heat from the fission products will continue to produce heat in the fuel elements
4. The fuel clad temperature must be maintained below the point where a cladding failure would occur.

The analysis showed that the maximum temperature that TRIGA fuel can tolerate without damage to the cladding and subsequent release of fission products is

1. 900°C for ***standard fuel with H/Zr =1.7***
2. 940°C for ***FLIP fuel with H/Zr =1.6***

The current design for the TRIGA 30/20 LEU fuel has an average H/Zr = 1.6 with a range of 1.57 to 1.65. Therefore, its temperature capability would be closer to the 940°C for the FLIP fuel that was analyzed in the 1979 SAR. Finally the maximum power density for the 30/20 LEU fuel calculated as part of this HEU/LEU conversion is 22.9 kW/rod which is within the bounds of the above analysis. Therefore, there are no changes to the LOCA analyses that are required for the conversion.

A stand-alone LOCA analysis was prepared for a response to a Request for Additional Information. The LOCA analysis is included in this SAR as Appendix LOCA: Analysis of Loss-of-Coolant Accident (LOCA).

13.5.3 Accidental Pulsing from Full Power

BOL, Beginning of Core Life LEU Core

The rapid insertion of a large amount of positive reactivity in the reactor operating at 1.0 MW is postulated. The method of inserting this reactivity is either by the ejection of an inserted transient rod or unplanned removal of a two-dollar experiment (an experiment that is required by Technical Specifications to be securely mounted in core). The Technical Specification also limits the maximum insertion of a transient rod to about that value which produces a maximum allowed fuel temperature no higher than the Safety Limit (1150°C for 30/20 LEU fuel). This reactivity is about $2.75 for the (30/20) core. Since the full worth of the transient rod is calculated to be $3.19, which could produce temperatures greater than the Safety Limit of 1150° C for 30/20 LEU fuel, accident analysis was performed assuming the full $3.19 value of the transient rod.

Pulsing from full reactor power (1.0 MW) would be clearly an accident. On the one hand, an administrative control prevents application of air to the piston for any reactor power above 1 kW. In addition, administrative procedures prevent the reactor operator from switching the Mode Switch to PULSE during high power operation and pushing the PULSE FIRE button. For the operator to do so clearly violates the two Safety Procedures.

The sequence of events leading to the postulated transient rod ejection accident at BOL is the following:

1. The reactor is in the steady-state mode, operating at 1.0 MW. The two redundant power scrams are set at 125% full power. The fuel temperature scram is set at 500 °C.
2. The transient rod is in its full down position, with a negative reactivity of about $3.19.
3. The operator turns the Mode Switch to PULSE.
4. The interlock that prevents energizing the pulse circuit when the reactor power is greater than 1 kilowatt fails.
5. The operator fires the transient rod, which reaches the maximum UP position in less than 0.1 second (the time for a full stroke travel).
6. When the reactor power rises from 100% power to the Power Scram set point of 125%, the reactor scrams with three control blades falling to their maximum inserted position in less than 2 seconds (the Technical Specification scram time for a full stroke).

The consequences of the above sequence of events are the following:

1. The reactor power increases from 1.0 MW to a peak pulsed power of about 1614 MW.
2. The maximum fuel temperature (997 °C) is reached immediately after the peak power.
3. The energy release is about 25.2 MW-sec in about 1.0 second when the maximum measured fuel temperature ($\hat{T}_{0.3}$ = 807 °C) is reached.
4. At peak fuel temperatures below 1150 °C, the strength of the clad maintains clad integrity so long as it remains water cooled.

This accident can be viewed in a number of ways. The transient rod reactivity is at the full stroke value of $3.19. The equilibrium pressure of hydrogen over the fuel is not achieved during the abbreviated pulse. It was incorrectly assumed in the earlier analyses that the back-fill air left in the fuel during manufacture is still present at EOL, whereas in fact it has disappeared, forming oxides and nitrides during the first days of operation at full power. With the action of the dual power scrams at 125%, the scram of the control rod bank starts even before peak pulsed power is attained.

The pulsing calculations from power have involved hand calculations using the Nordheim -Fuch model since BLOOST cannot handle a pulse from power. BLOOST is a zero-dimensional, combined reactor kinetics-heat transfer code. It cannot handle the "inverted U" temperature distribution in fuel operating in steady state coupled with the "U shaped" power distribution in a pulsed fuel element. BLOOST calculates an average core temperature as a function of time.

Calculations were made to establish the average fuel temperature at the steady state starting power of 1.0 MW. A value of 306°C was determined for $\overline{T}_{core}$.

The BLOOST calculations indicate that the highest average fuel temperature in the pulsed core immediately after the transient pulse is 492°C. From three-dimensional diffusion theory calculations, the peak-to-average power ratio was determined for steady state operation to be 3.8. Although the highest temperatures occur at the center of the hottest fuel element during 1 MW steady state operation (500°C) and before the pulse, the maximum fuel temperature after pulsing occurs at the edge because of the large power peak.

The peak-to-average value at the edge of the hottest pulsed element is 3.8. Using these power ratios and considering the energy release during the transient superimposed on the energy density levels under steady state, coupled with the volumetric heat content of the fuel, a maximum fuel temperature of 997°C was obtained based on the average core temperatures computed by BLOOST.

An alternative method of producing the accidental pulse from full power is to remove an installed two-dollar experiment. Although the Technical Specification requires such an experiment to be securely locked in position in the core, somehow, the experiment is loosened and yanked up out of the core. Because of its mass, the removal time is typically assumed to be 0.3 seconds (considerably longer than the 0.1 second for the engineered transient rod drive). The much slower withdrawal time will result in activating the 125% power scrams at a lower portion of the reactivity insertion curve. This will result in lower peak power and lower peak fuel temperatures than those for a $3.19 transient rod. Thus, neither accident endangers the reactor.

To review, the accidental pulsing of the transient rod requires the following:

1. Failure of the 1 kW interlock that prevents air from being applied to the transient rod piston for reactor power above 1 kW.

2. Failure of reactor operator to follow written procedures.

Note: Both must occur for the accident to occur.

The accidental removal of a two-dollar experiment requires the following:

1. Deliberate violation of written procedures as an operator unfastens the two-dollar experiment and proceeds to yank it out of the core.

Note: Only one gross failure must occur for this accident to occur – no interlocks need to fail.

EOL Pulsing from Full Power at 1000 MWD Burnup

The BLOOST code was run with input appropriate for the LEU core at 1000 MWD. Pulsing results were obtained for a series of reactivity insertions under the same conditions as set forth in the above section. Table 31 lists these results along with those for the beginning of core life.

Table 32 Pulsing from 1.0 MW, Beginning of Life (BOL) and End of Life (EOL)

Parameter	BOL			EOL (1000 MWD)		
$\Delta k/k\beta$ ($)	2.00	2.75	3.19	2.00	2.75	3.19
$\hat{T}$ (°C)	670	925	997	650	930	1003
$\bar{T}_{core}$ (°C)	384	470	492	308	417	443
$\hat{T}_{0.3}$ (°C) –D6NW	638	762	798	798	637	675
$\hat{T}_{0.3}$ (°C) –C4NW	638	769	807	538	690	731
MW-sec	12.4	22.1	25.2	12.9	24.6	28.2
$\hat{P}$ (MW)	388	1239	1614	371	1350	1767

In Table 32, the effects are evident for shifts in the power distribution with burnup for the core. The measured temperature ($\hat{T}_{0.3}$) at $2.00 pulse decreases from 638°C to 498°C with burnup. The peak temperature in the hottest fuel element rises slightly, from 997°C to 1003°C; but even 1003°C is below the 1150°C Safety Limit.

The conclusion is that accidental pulsing from full power is not a hazard for the reactor, either at Beginning- or End-of 1000 MWD burnup.

14.0 Revised Technical Specifications

A revision of the Technical Specifications from the WSU License No. R-76, through Amendment No. 18, dated August 26, 2004 is included with this submittal under a separate cover. The Technical Specifications are submitted along with this SAR under a separate cover to maintain internal consistency with the Technical Specification Amendment No. 18 with respect to pagination and section heading numbers. Changes from Amendment No. 18 are indicated with change bars.

Appendix A

A.1. LEU (30/20) Startup Plan

A.1.1 Initial Criticality

Based on practical experience derived from the criticality approach with several other TRIGA reactors, criticality is expected with a loading of 58-68 fresh, 30/20 fuel elements. TRIGA 30/20 fuel is defined as 30 wt.% U, 19.75% enriched in U-235.

The loading of fuel elements to obtain criticality will be accomplished using the standard inverse multiplication curve (1/M) approach. This is based on the fact that subcritical multiplication is given as

$$M = 1/(1-k)$$

from which one obtains

$$1/M = 1-k$$

where k ranges from 0 (no fuel) to 1 (at criticality). The experimental values for 1/M subcritical multiplication are given by the count rate with no fuel, C_0, divided by C_n for loading step n. However, for the present 1/M application for approach to critical, the value C_0 can start at any convenient loading point. For the TRIGA application, C_0 is usually the count rate with the instrumented fuel elements installed in the core together with the fuel in the 3-rod clusters that contain the transient rod, but the transient rod withdrawn from the core.

<u>Acceptance Criteria</u>: The 1/M criticality is expected with a fuel loading as follows:

Partially Burned 8.5/20 SFEs from 24 to 32 rods
Fresh 30/20 SFEs from 43 to 47 rods

<u>Contingency for Failure to Meet Acceptance Criteria</u>

A failure of agreement of data generated by the inverse multiplication/critical loading test with the predicted number of fuel elements would suggest that either the inverse multiplication data are erroneous, or the predictive model is inaccurate. Resolution will be made by reexamining fuel loading procedures, instrument readings, data and calculations, and the predictive model.

A.1.2 Critical Mass and Criticality Conditions for the 30/20 LEU Core

Measurements Upon Attaining Criticality

The core excess reactivity shall be determined upon reaching criticality with all control rods fully withdrawn, using the period method.

The estimated control rod reactivity worth for each control rod is obtained using the Rod Drop technique with either the Reactivity Computer or the classical rod drop negative period measurements with a stopwatch.

Acceptance Criteria: The Rod Drop reactivity worths for the scrammable control rods are expected to lie between $0.40 and $3.20.

Contingency for Failure to Meet Acceptance Criteria

The reactivity worths for the scrammable control rods will be dependent upon the identity of the control rod, and core configuration at the time that the reactivity worths are determined. It is likely that the worth of control blade number five (the non-scrammable stainless steel regulating blade) will be less than $0.40. If the measured worth of a scrammable control rod or blade falls outside the Acceptance Criteria the deviation will be recorded, the cause determined, and will be reported to the Reactor Safeguards Committee. The WSU Facility Director (a licensed Senior Reactor Operator) and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications. The reactor will be shut-down and secured, and use of the reactor, including movement of fuel into or out of the reactor or maintenance of reactor control systems will not be permitted until the WSU Facility Director, Reactor Supervisor, General Atomics, and the Reactor Safeguards Committee agree that continuation of refueling activities may be done safely.

A.1.3 Initial Control Rod Calibration Tests

The WSU 1 MW TRIGA reactor with Mixed LEU core contains three (3) shim control blades, one (1) Servo control blade and one (1) water-followed transient rod.

The reactivity insertion procedure is used in conjunction with the Reactivity Computer to calibrate each control rod as a function of its withdrawal distance from the fully inserted position in the core. For this procedure, the available core excess reactivity must at least equal the reactivity worth of the most reactive control rod. Therefore, additional reactor fuel must be added to provide the required core excess reactivity.

Each control rod is calibrated starting from its fully inserted position. Each small positive reactivity insertion is indicated on the reactivity computer and is then counter balanced by an appropriate insertion of negative reactivity from the remaining control rods operating in a bank.

A differential and integral calibration curve is prepared for each control rod. Using the calibration curves for the control rods, determine a reliable value for the interim core excess reactivity with the control rods in a banked configuration.

Acceptance Criteria: The calibration curve results for control rod worth are expected to vary between a low value of approximately $0.20 and a high value of $3.00 to $4.00, depending on the type of rod and location in the core.

Contingency for Failure to Meet Acceptance Criteria

If the measured worth of a control rod or blade falls outside the Acceptance Criteria the deviation will recorded, the cause determined, and will be reported to the Reactor Safeguards Committee. The WSU Facility Director (a licensed Senior Reactor Operator) and Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications. The reactor will be shut-down and secured, and use of the reactor, including movement of fuel into or out of the reactor or maintenance of reactor control systems will not be permitted until the WSU Facility Director, the Reactor Supervisor, General Atomics, and the Reactor Safeguards Committee agree that continuation of refueling activities may be done safely.

A.1.4 Final Core Loading/Final Rod Calibrations

The required fuel loading for achieving full power operation can be installed. A total of 119 fuel elements, 51 – Fresh 30/20 LEU SFEs and 68 Burned 8.5/20 LEU SFEs will be loaded. While loading fuel, all but two control rods are full down. If more than four fuel elements have been added, it is necessary to recalibrate individually all control rods using the procedure already described above in Section A.1.3.

After the final core loading is complete, and before additional control rod calibrations, it is useful to establish an initial setting of the console reactor power using the temperature coefficient of reactivity, α. For a TRIGA reactor, this leads to a value of about "1 cent reactivity per kilowatt of power". This relationship holds within a factor of about two (2) for all TRIGA reactors with reactor power levels up to, and in excess of, 100 kW and can be used initially to make approximate power level settings.

Following a final recalibration of all control rods, the excess reactivity with the cold, clean fuel is determined for a full core loading. Additional measurements will be performed to assure that the "stuck rod criteria" is met by the assembly of control rods (i.e., reactor shut down by at least 25 cents reactivity with the most reactive control rod fully removed from the core).

At this point, the "zero power" reactivity commissioning tests are complete and the reactor is ready for the calorimetric power tests.

Acceptance Criteria:

- With 119 fuel elements the excess reactivity is expected to be about \$ 6.37, the computed value.
- With the same core and same location, the "shut down margin" the shall be greater than \$0.25 with the most reactive scrammable rod and the stainless steel regulating blade in the fully withdrawn positions.

Contingency for Failure to Meet Acceptance Criteria

If the excess reactivity value is not within the range \$5.87 - \$6.87 the deviation will recorded, the cause determined, and will be reported to the Reactor Safeguards Committee. The WSU Facility Director (a licensed Senior Reactor Operator) and Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications. The reactor will be shut-down and secured, and use of the reactor, including movement of fuel into or out of the reactor or maintenance of reactor control systems will not be permitted until the WSU Facility Director, the Reactor Supervisor, General Atomics, and the Reactor Safeguards Committee agree that continuation of refueling activities may be done safely.

The minimum shut down margin shall be \$0.25, with the most reactive scrammable control rod and the stainless steel regulating rod fully withdrawn from the core. If this Acceptance Criterion is not met the reactor shall be shut down and secured. The reactor shall not be operated with a shutdown margin that is less than \$0.25. If it is determined that the shutdown margin is less than \$0.25 the WSU Facility Director and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards committee to determine an appropriate remedial course of action. Possible remedial actions include removing fuel or reflector from the reactor until a determination is made of cause and remedy of the smaller than expected shutdown margin.

A.1.5 Calorimetric Reactor Power Calibration

The calorimetric power calibration takes advantage of the fact that natural convection provides adequate cooling for a TRIGA core operating at power levels up to and including 2.0 MW.

In the so-called "slope" method of calibration, the rate of temperature rise will be determined for the reactor pool water [dT/dt (°C/hr)] while the reactor is operating at power P and the tank water is stirred. For the WSU TRIGA reactor, the so-called Tank Constant (°C /MWh) is calculated from the water volume in the reactor tank. From this and the measured time rate of pool water temperature rise, the actual reactor power can be computed as

$$P(MW) = [dT/dt\ (°C/hr) / \text{Tank Constant}\ (°C/MWh)]$$

The calorimetric power calibration with effective circulation of tank water is conducted in two steps. The first is conducted at low to intermediate power (~250 kW) to determine the initial, nearly correct power reading on all power channel detectors. The second power calibration will then be performed at an indicated power level of about 750 kW, close to the licensed reactor power of 1 MW.

With the power level P computed from the above formula, and with the reactor operating at this power, the detectors for the power measuring channels on the reactor console will be adjusted to assure that the console correctly indicates this power level.

Note: At this point, the low-to-intermediate power tests for commissioning have been completed. Tests at higher, and full, power can now be conducted.

Acceptance Criteria: After the final power calibration, all power channel indications will agree within 2% at full reactor power, 1.0 MW.

Contingency for Failure to Meet Acceptance Criteria

If the power channel indications do not agree within 2% at full reactor power (1 MW) the power calibration will be repeated.

If the power channel indications do not agree within 2% after a second calibration, the WSU Facility Director and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to determine an appropriate course of action.

A.1.6 Initial Approach to Full Power

Outline of Approach

The object of this test is to approach full power operation in carefully programmed steps, recording fuel temperatures, all power indications on each measuring channel, and all control rod positions together with calculated reactor core excess reactivity. Continue the stepwise power increase until the power level of 1.0 MW is reached.

It is important to complete the stepwise increase in power without any decreases in reactor power so that the expected increase in measured fuel temperatures can be quantified (i.e., future values of fuel temperatures at power levels below 1.0 MW will be slightly increased above the very first measured values). During the first operation at 1.0 MW, the hot fuel will expand, stretching the fuel cladding by a small, permanent amount. For the second, and subsequent approaches to full power (1 MW), the fuel must heat to a slightly higher temperature to cause expansion to the slightly larger cladding diameter. The two sets of measured fuel temperatures demonstrate this effect.

Repeat the stepwise increase in power starting from a low power (< 1 kW). Record the fuel temperatures at each of the same power levels used in the previous stepwise rise in power. Plot the two sets of fuel temperatures versus reactor power to demonstrate the "hysteresis" effect caused by peak fuel temperature.

Acceptance Criteria: At full power (1.0 MW), the reactivity loss is expected to lie in the range from \$2.25 to \$2.75, values that verify the presence of a large negative coefficient of reactivity.

Contingency for Failure to Meet Acceptance Criteria

The observed negative temperature coefficient of reactivity is expected to be bracketed by the \$2.25 to \$2.75 range. If the observed value lies outside the Acceptance Criteria range the WSU Facility Director (a licensed Senior Reactor Operator) and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications.

Linearity Check on the Power Indication Channels

For several user applications of the research reactor, it is useful to be able to rely on the linearity of the power readout instrumentation on the reactor console. To establish the degree of linearity for this power instrumentation, a test is conducted using as a standard the well established linearity of the current in the fission detector with reactor power level (for currents above the dark current ~ 5×10^{-8} amp). This D.C. current (up to about 1.0 milliamp at 1.0 MW in steady state) provides an adequate reference against which to compare the console power indications over most of the important energy range.

Take data from low power (few hundred watts) up to 1.0 MW for each power measuring console channel. Prepare a log-log graph for each power channel showing the console power indication versus the D.C. return current from a fission counter detector. The straightness of the resulting line connecting the data points demonstrates the linearity of the console power measuring channels.

Acceptance Criteria: A log-log plot of the detector indications versus the D.C. return current in a fission counter shall be a nearly straight line over a power span from about 100 kW to 1.0 MW, thus demonstrating detector channel linearity.

Contingency for Failure to Meet Acceptance Criteria

Correct functioning of the log power channel is required as one of the minimum number of measuring channels as a Limiting Condition of Operation.

Indications of lack of accurate detector performance shall be investigated and appropriate corrective action taken, e.g. repair or replacement of malfunctioning components. The reactor may only be operated when the log power channel is operating correctly.

Tests of 125% Power Scram

The power level Scram at 125% of 1.0 MW is an important component of the Safety System. Operation at about 1.0 MW with Scram at 1.25 MW assures an adequate margin of safety. Scram at 1.25 MW is sufficiently above the full power (1.0 MW) that normal operational variation around 1.0 MW is unlikely to accidentally activate the 125% scram point.

The object of the test is to assure that a power level of 125% of 1.0 MW will in fact scram the reactor. At this point in the Commissioning Program, all 125% scram checks have been performed electronically with low or zero reactor power.

Acceptance Criteria: The reactor shall scram reliably when a Safety power channel reaches an indicated 1.25 MW.

Contingency for Failure to Meet Acceptance Criteria

Correct functioning of the high power scram on the Safety Channel is a Limiting Condition of Operation. The reactor shall not be operated unless the high power scram operates correctly.

A.1.7 Pulsing Mode of Operation

Criteria for Determining Maximum Reactivity Insertion (Maximum Pulsed Energy Release)

Several considerations are at work in determining the maximum pulse power/reactivity insertion:

(i) Determine maximum reactivity insertion that produces maximum permitted fuel temperature in hottest fuel rod;
(ii) Determine if value of maximum reactivity insertion decreases as prompt negative coefficient of reactivity decreases with fuel burnup.
(iii) Determine reduced value of maximum reactivity insertion as long term steady state operation creates increased ratio of H/Zr in outer periphery of fuel rods.
(iv) If longest core lifetime (burnup) is desired, limit pulsed $\hat{T}$ in hottest fuel to a value no higher than $\hat{T}$ in hottest fuel in steady state mode of operation.
(v) Recognize the experimenters' desire for peak thermal neutron flux; hence, largest safe reactivity insertion.

The reactor operator/owner must balance the long term needs of the WSU facility against the users' requirements as an aid in determining the maximum permitted reactivity insertion. The reactor operator/owner must also establish whether the peak pulsed fuel temperature in the hottest fuel rod will be restricted to values (1) no greater than the peak fuel temperature in steady state operation for longest core life, (2) up to the safe temperature limit set forth in the applicable SAR; or (3) somewhere in between these limits.

Pulse Calibration Procedures

- Install a high speed analog or digital recorder to record the peak power (nv) output from the pulsing channel, one or more fuel temperatures, and an accurate shape of the pulse. The use of the (nv) data will permit an accurate evaluation of the peak power; and the prompt reactor period, which, with the prompt neutron lifetime, can be used to determine the effective pulsed reactivity insertion. Provide separate calibration plots of peak power and fuel temperature.
- Perform a series of pulses starting at about \$1.25 and increasing in 25 cent increments to the maximum reactivity determined from the considerations set forth above. For each pulse, record the high speed data for $\hat{P}$, the initial pulsed reactor period deduced from the plot of data, and the time variation of the fuel temperature(s) in the hottest fuel element. For at least one large pulse, record the peak fuel temperature before allowing any rod to scram for several seconds after the pulse.
  Note: If it turns out not to be possible to record the peak power simultaneously on two high speed recorder channels having different gains, it may be necessary to make at least two pulses at each reactivity insertion, one with gain set to give $\hat{P}$ and one with gain set to give proper period data early in the rise of the pulse.
  (See Acceptance Criteria in Section 10.5.7.3 below)

Pulsing Data Report

Calculate the effective reactivity insertion for each pulse from the measured prompt reactor period.

For the range of pulse insertions, plot $\hat{P}$ versus (reactivity insertion)^2; 1/period versus reactivity insertion; fuel temperature(s) versus reactivity insertion; and Integrated Energy (MW-sec) versus reactivity insertions.

Acceptance Criteria: Plots of energy release and peak power for pulsing performance shall be consistent with a linear (straight line) dependence of either (Δk_P) or $(\Delta k_P)^2$, as appropriate.

Contingency for Failure to Meet Acceptance Criteria

Plots of energy release and peak power for pulsing performance will be compared with similar plots prepared from pulsing data previously generated at the WSU reactor. Historical pulsing data will be used to establish estimated uncertainty in measured values for energy release and peak power. The WSU Facility Director and the Reactor Supervisor will consult with General Atomics and the Reactor Safeguards Committee to assess operational and safety implications if plots of energy release and peak power for pulsing performance do not follow the expected linear trends, within experimental uncertainty.

Appendix B

B.2. FLIP (HEU) and LEU (30/20) Fuel Storage

In principle, from the criticality point of view, TRIGA fuel of any type can be stored in the same facility.

B.2.1 WSU Fuel Storage Facilities

Figure 40 shows the MCNP model for the dry storage of fresh, unirradiated fuel. There are five dry tubes (4 inches diameter) made of aluminum. Each tube can store two fuel elements, and they are spaced 3 inches apart.

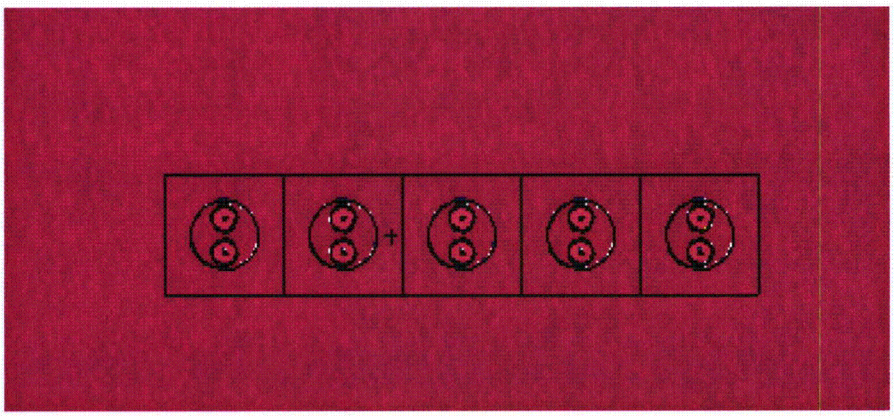

Figure 40 MCNP X-Y Plot for the Dry Fuel Storage Tubes

Figure 41 illustrates the in-tank wet racks storage facility. This consists of four large storage racks free standing on the pool floor, and the racks are spaced 8 inches apart. Each rack has nine pits, and each pit can store 4-rods cluster.

Currently, the in-tank storage has free space to store 32, 4-rod fuel clusters. The wet racks have 0.75 inch spacing between adjacent clusters, and 8.75 inches spacing along the width.

The approach to storage has been very conservative, with the aim to establish upper bounds for criticality.

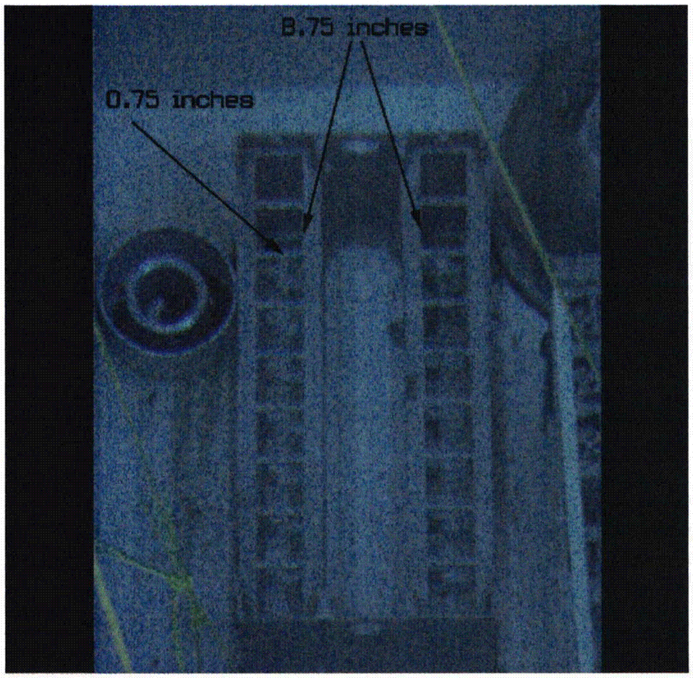

Figure 41 In-Tank Fuel Storage Arrangement

For the dry, fresh fuel storage, the MCNP model included the aluminum tubes with two fuel elements inside. The MCNP program was run twice, once with air in the storage room, and once with the storage room flooded.

Storage of LEU (30/20) in air	Storage of LEU (30/20) flooded
$k_{eff} = 0.03279 \pm 0.00006$	$k_{eff} = 0.37572 \pm 0.00028$

The MCNP model for the in-tank storage was similarly conservative for this 4x9 stand alone floor storage facility. In the MCNP model 4x9 racks were modeled, and each pit was filled with the four fresh FLIP or four fresh 30/20 LEU fuels. As expected, the results for both FLIP and LEU (30/20) fuel gave very small values of k_{eff}, far below the limit of 0.8.

	FLIP	LEU
4x9, water reflected	$k_{eff} = 0.63456 \pm 0.00035$	$k_{eff} = 0.64886 \pm 0.00035$

Conclusion: The storage of fresh LEU (30/20) fuel in the fresh fuel locker is entirely safe, even if the locker were to flood.
The storage of spent FLIP fuel and/or fresh LEU (30/20) fuel in the wet storage facilities is entirely safe.

Appendix MHA

Analysis of Maximum Hypothetical Accident

The Maximum Hypothetical Accident (MHA) for a TRIGA reactor is defined as the loss of the integrity of the fuel cladding of one fuel rod in air. This MHA definition is consistent with the Design Basis Accident (DBA) defined in NUREG/CR-2387 (Ref. 13).

To compare the relative abundance of fission products, the pertinent operating parameters are compared for a FLIP core and a LEU (30/20) core. The major parameter important to radiological impact is the fission product inventory due to burnup and power density. The WSU mixed HEU Core 34A has a life of about 1000 MWD and is essentially the same as the WSU mixed LEU Core 35A as discussed in Section 4.5.6. The energy burnup capability for FLIP fuel is 77 MWD which is much greater than 50% burnup. For WSU LEU (30/20) fuel, the energy burnup capability based on 50% of the U-235 gives 54 MWD/fuel element (NUREG-1282).

On a per MWD basis, the fission product inventory for TRIGA HEU fuel and LEU fuel are negligibly different. For TRIGA HEU fuel, less than 1% of the fission power at EOL has come from Pu-239. However, for TRIGA LEU fuel, the percentage of fission power from Pu-239 increases to less than 4%. In both fuels, almost all of the fission product inventory is due to U-235 fission. Any inventory difference between HEU and LEU fuel for a given burnup is minor compared to other uncertainties in the assessment of radiological dose.

The fission product inventories in Table 33 were calculated using ORIGEN. Both FLIP and LEU (30/20) inventories were calculated for a single fuel element. The element was assumed to have a conservative power density of 28 kW. The burnup calculation was performed for 200 days (5.6 MWD) in order to achieve saturation levels for all of the isotopes except Kr-85. Maximum inventories occurred after 200 MWD of burnup which maximized buildup while minimizing the effect of fissile material depletion. The burnup calculation for Kr-85 was extended to 77 MWD for FLIP fuel and 54 MWD for LEU (30/20) fuel.

The rate at which fission products are removed from the pool room and released into the environment during an MHA depends on the removal rate of pool room air by the ventilation system. In the normal operation mode, air is exhausted from the pool room at the rate of 4500 cfm (2.12×10^6 cm^3/sec). On detection of high radiation levels by the Continuous Air Monitor (CAM), the HVAC switches to dilution mode. In the dilution mode, 300 cfm (1.41×10^5 cm^3/sec) of air from the pool room is passed through a HEPA filter system, diluted with 1700 cfm of outside air and discharged to the atmosphere. In the dilution mode, 2000 cfm (9.44×10^5 cm^3/sec) of air is discharged with a dilution factor of 6.67. In the dilution mode, the HEPA filter would remove at least 90% of the iodine and particulates from the exhaust air. If the ventilation system is switched to isolation mode, the release to the environment would only be by leakage from a sealed building which is estimated to be of the order of 100 cfm (4.72×10^4 cm^3/sec). The analysis of the MHA assumes that the ventilation is automatically switched to the dilution mode but that the HEPA filter is ineffective in removing volatile fission products from the exhaust air.

Table 33
Gaseous Fission Products in a Single TRIGA Fuel Rod (28 kW/rod)

Isotope	FLIP (Curies)	LEU (30/20) (Curies)	Half-Life
Br-82	1	1	35.3 hr
Br-83	127	126	2.3 hr
Br-84	246	245	31.8 min
Br-85	291	288	3.0 min
I-131	681	685	8.1 days
I-132	1022	1029	2.3 hr
I-133	1587	1590	21.0 hr
I-134	1829	1829	54.0 min
I-135	1488	1488	6.8 hr
Kr-83m	127	126	1.9 hr
Kr-85m	290	288	4.4 hr
Kr-85	23	18	10.7 yr
Kr-87	594	589	78.0 min
Kr-88	840	834	2.8 hr
Kr-89	1095	1084	3.2 min
Xe-131m	7.5	7.5	12.0 days
Xe-133m	46	46	2.3 days
Xe-133	1547	1550	5.3 days
Xe-135m	275	276	15.0 min
Xe-135	962	1057	9.0 hr
Xe-137	1449	1449	3.9 min
Xe-138	1504	1504	17.0 min

A fuel element with a conservative maximum temperature of 535° C has an average fission product release fraction of 2.6×10^{-5} into the clad gap. This release fraction is averaged over the fuel volume and temperature. Evacuation drills have demonstrated that personnel within the pool room can be evacuated within 5 minutes. The 5 minute occupational exposure during an MHA in which a single LEU 30/20 fuel rod fails in air is affected by radioactive decay and pool room ventilation during the 5 minute exposure. The time-integrated concentration during the 5-minute exposure is determined by the following equation:

$$\bar{C}(0 - 300\,s) = \frac{A}{V(\lambda_L + \lambda_D)}\left[1 - e^{-(\lambda_L + \lambda_D)300\,s}\right]$$

where A = activity released into pool room,
V = pool room volume,
λ_L = pool room ventilation rate = 1.41×10^{-4} sec^{-1},
λ_D = radioactive decay rate, sec^{-1}.

The fission product release from a single LEU 30/20 fuel element into the pool room atmosphere, resulting time-averaged concentration, and whole body dose are presented in Table 34. The release is instantaneous into the pool room which has a volume of 1×10^9 cm^3. The thyroid dose is presented in Table 35. The whole body and thyroid dose rates are calculated using dose conversion factors (DCFs) from EPA-400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. These DCFs include both inhalation and external exposure effects. The 21.5 mrem whole body and 2.77 rem thyroid which are much less than the regulatory limits stated in NUREG-1537 of 5 rem whole body and 30 rem thyroid for research reactors licensed before January 1, 1994.

Table 34

Pool Room Fission Product Concentrations and Associated 5-Minute Whole Body Worker Dose for Single LEU 30/20 Fuel Rod Cladding Failure in Air

Isotope	Release to Pool Room (mCi)	Time-Integrated Concentration (μCi-s/cm^3)	DCF in rem per μCi/cm^3/hr	Whole Body Dose (mrem)
Br-82	0.026	7.63×10^{-6}	1250	2.6×10^{-3}
Br-83	3.3	9.50×10^{-4}	83	0.02
Br-84	6.4	1.77×10^{-3}	125	0.06
Br-85	7.5	1.31×10^{-3}	115	0.04
I-131	17.8	5.23×10^{-3}	220	0.32
I-132	26.8	7.76×10^{-3}	1400	3.02
I-133	41.3	1.21×10^{-2}	350	1.18
I-134	47.6	1.35×10^{-2}	1600	6.01
I-135	38.7	1.13×10^{-2}	950	2.99
Kr-83m	3.3	9.48×10^{-4}	100	0.03
Kr-85m	7.5	2.19×10^{-3}	93	0.06
Kr-85	0.5	1.37×10^{-4}	1.3	5.0×10^{-5}
Kr-87	15.3	4.40×10^{-3}	510	0.62
Kr-88	21.7	6.30×10^{-3}	1300	2.28
Kr-89	28.2	5.08×10^{-3}	1200	1.69
Xe-131m	0.2	5.73×10^{-5}	4.9	7.8×10^{-5}
Xe-133m	1.2	3.51×10^{-4}	17	1.7×10^{-3}
Xe-133	40.3	1.18×10^{-2}	140	0.46
Xe-135m	7.2	1.88×10^{-3}	250	0.13
Xe-135	27.5	8.05×10^{-3}	140	0.31
Xe-137	37.7	7.35×10^{-3}	110	0.22
Xe-138	39.1	1.04×10^{-2}	710	2.05
Total Whole Body Dose				21.50

Table 35

Pool Room Fission Product Concentrations and Associated 5-Minute Thyroid Worker Dose for Single LEU 30/20 Fuel Rod Cladding Failure in Air

Isotope	Release to Pool Room (mCi)	Time-Integrated Concentration (μCi-s/cm^3)	DCF in rem per μCi/cm^3/hr	Thyroid Dose (rem)
I-131	17.8	5.23×10^{-3}	1.3×10^{6}	1.89
I-132	26.8	7.76×10^{-3}	7.7×10^{3}	0.02
I-133	41.3	1.21×10^{-2}	2.2×10^{5}	0.74
I-134	47.6	1.35×10^{-2}	1.3×10^{3}	4.9×10^{-3}
I-135	38.7	1.13×10^{-2}	3.8×10^{4}	0.12
Total Thyroid Dose				2.77

The gaseous radioactive material discharged from the facility ventilation system will be diluted by atmospheric air in the lee of the building due to turbulent wake effects. The nearest member of the public could be as close as 50 ft (15 m) which would place the individual within the 20-m wake cavity formed by the building. Using a Gaussian plume dispersion model, the initial dimensions of the plume are defined by the following parameters (NUREG/CR-4691, 1990):

$$\sigma_y = W_b/4.3$$

$$\sigma_z = H_b/2.15$$

where W_b and H_b are the width and height of the building.

Using a conservative windspeed, u, of 1 m/sec (2.2 mph), a building height of 8.53 m (28 ft) and a minimum building width of 17.07 m (56 ft), the χ/Q for the nearest member of the public is calculated as follows:

$$\frac{\chi}{Q} = \frac{1}{\pi \sigma_y \sigma_z u} = 0.0202 \ s/m^3$$

The nearest member of the public is assumed to remain throughout the accidental release. The fraction of the activity released to the pool room that is discharged to the environment depends on the radioactive decay rate and the pool room exhaust rate while in dilution mode. No credit is taken for deposition mechanisms either within the pool room or in the environment. The fraction of activity discharged to the environment is calculated by the following relationship:

$$\frac{\lambda_L}{\lambda_L + \lambda_D}$$

The ventilation system will have switched to dilution mode during the MHA but no credit is taken for the HEPA filter in reducing iodine release. Therefore the fractional discharge rate from the pool room is 1.41 x 10^{-4} per second or 8.46 x 10^{-3} per minute. The resulting environmental release, time-integrated concentration and dose to the whole body and thyroid are presented in Tables 36 and 37. These doses are much less than the 0.5 rem whole body and 3 rem thyroid doses acceptable for members of the public for research reactors licensed before January 1, 1994.

Table 36
Environmental Fission Product Release and Whole Body Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Pool Room (mCi)	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m³)	DCF in rem per μCi/cm³/hr	W.B. Dose (mrem)
Br-82	0.026	0.025	0.51	1250	1.8×10^{-4}
Br-83	3.3	2.06	41.52	83	9.6×10^{-4}
Br-84	6.4	1.78	35.98	125	0.001
Br-85	7.5	0.26	5.34	115	1.7×10^{-4}
I-131	17.8	17.69	357.25	220	0.022
I-132	26.8	16.79	339.10	1400	0.132
I-133	41.3	38.82	784.08	350	0.076
I-134	47.6	18.89	381.60	1600	0.170
I-135	38.7	32.22	650.81	950	0.172
Kr-83m	3.3	1.91	38.50	100	0.001
Kr-85m	7.5	5.72	115.43	93	0.003
Kr-85	0.5	0.47	9.45	1.3	3.4×10^{-6}
Kr-87	15.3	7.47	150.87	510	0.021
Kr-88	21.7	14.58	294.43	1300	0.106
Kr-89	28.2	1.06	21.40	1200	0.007
Xe-131m	0.2	0.19	3.92	4.9	5.3×10^{-6}
Xe-133m	1.2	1.17	23.58	17	1.1×10^{-4}
Xe-133	40.3	39.87	805.41	140	0.031
Xe-135m	7.2	1.11	22.43	250	0.002
Xe-135	27.5	23.86	482.00	140	0.019
Xe-137	37.7	1.71	34.58	110	0.001
Xe-138	39.1	6.72	135.73	710	0.027
Total Whole Body Dose					0.792

Table 37
Environmental Fission Product Release and Thyroid Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Pool Room (mCi)	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m³)	DCF in rem per μCi/cm³/hr	Thyroid Dose (mrem)
I-131	17.8	17.69	357.25	1.3×10^{6}	129.01
I-132	26.8	16.79	339.10	7.7×10^{3}	0.73
I-133	41.3	38.82	784.08	2.2×10^{5}	47.92
I-134	47.6	18.89	381.60	1.3×10^{3}	0.14
I-135	38.7	32.22	650.81	3.8×10^{4}	6.87
Total Thyroid Dose					184.66

The nearest resident to the WSU reactor is at the Valley Crest Village apartments located approximately 600 m (2000 ft) southwest of the facility. Conservative weather conditions of Stability Class F and windspeed of 1 m/s (2.2 mph) are assumed to persist throughout the release from the MHA. Deposition processes and plume meander are neglected. The environmental release is the same for the nearest resident as for the nearest member of the public given in Tables 36 and 37. The atmospheric dispersion parameters, σ_y and σ_z, based on the methodology of NUREG/CR-1641 and DOE/TIC-11223 at 600 m are 27.1 and 11.0 m, respectively. The resulting χ/Q for the nearest resident is 1.07×10^{-3} s/m^3. The environmental release, time-integrated concentration and dose to the whole body and thyroid are presented in Tables 38 and 39. These doses are much less than the 0.5 rem whole body and 3 rem thyroid doses acceptable for members of the public for research reactors licensed before January 1, 1994.

Table 38
Environmental Fission Product Release and Whole Body Exposure for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m^3)	DCF in rem per μCi/cm^3/hr	Whole Body Dose (mrem)
Br-82	0.025	0.03	1250	9.3×10^{-6}
Br-83	2.06	2.20	83	5.1×10^{-5}
Br-84	1.78	1.91	125	6.6×10^{-5}
Br-85	0.26	0.28	115	9.0×10^{-6}
I-131	17.69	18.92	220	0.001
I-132	16.79	17.96	1400	0.007
I-133	38.82	41.53	350	0.004
I-134	18.89	20.21	1600	0.009
I-135	32.22	34.47	950	0.009
Kr-83m	1.91	2.04	100	5.7×10^{-5}
Kr-85m	5.72	6.11	93	1.6×10^{-4}
Kr-85	0.47	0.50	1.3	1.8×10^{-7}
Kr-87	7.47	7.99	510	0.001
Kr-88	14.58	15.60	1300	0.006
Kr-89	1.06	1.13	1200	3.8×10^{-4}
Xe-131m	0.19	0.21	4.9	2.8×10^{-7}
Xe-133m	1.17	1.25	17	5.9×10^{-6}
Xe-133	39.87	42.66	140	0.002
Xe-135m	1.11	1.19	250	8.3×10^{-5}
Xe-135	23.86	25.53	140	9.9×10^{-4}
Xe-137	1.71	1.83	110	0.006
Xe-138	6.72	7.19	710	0.001
Total Whole Body Dose				0.042

Table 39
Environmental Fission Product Release and Thyroid Exposure
for Single LEU 30/20 Fuel Rod Failure in Air

Isotope	Release to Environment (mCi)	Time Integrated Concentration (μCi-s/m^3)	DCF in rem per μCi/cm^3/hr	Thyroid Dose (mrem)
I-131	17.69	18.92	1.3×10^6	6.83
I-132	16.79	17.96	7.7×10^3	0.04
I-133	38.82	41.53	2.2×10^5	2.54
I-134	18.89	20.21	1.3×10^3	0.01
I-135	32.22	34.47	3.8×10^4	0.36
Total Thyroid Dose				9.78

"MELCOR Accident Consequence Code System (MACCS): Model Description," NUREG/CR-1691, Vol. 2, February 1990.

Hanna, Steven R., Gary A. Briggs, and Rayford P. Hosker, Jr., "Handbook on Atmospheric Dispersion," DOE/TIC-11223, 1982.

Appendix LOCA

Analysis of Loss-of-Coolant Accident (LOCA)

The strength of the fuel element clad is a function of its temperature. The stress imposed on the clad is a function of the fuel temperature as well as the hydrogen-to-zirconium ratio, the fuel burnup, and the free gas volume within the element. The analysis of the stress imposed on the clad and strength of the clad uses the following assumptions:

1) The fuel and clad are at the same temperature.
2) The hydrogen-to-zirconium ratio is 1.7 for standard fuel (8.5 wt%) and 1.6 for FLIP fuel and 30/20 fuel.
3) A space one-eighth inch high within the clad represents the free volume within the element.
4) The reactor contains fuel that has undergone burnup equivalent to 77 MW-days for FLIP fuel and 54 MW-days for LEU 30/20 fuel.
5) Maximum operating temperature of the fuel is 600° C.

The fuel element internal pressure P is given by:

$$P = P_h + P_{fp} + P_{air}$$

where:

P_h is the hydrogen pressure;

P_{fp} is the pressure exerted by volatile fission products; and

P_{air} is the pressure exerted by trapped air.

For hydrogen-to-zirconium ratios greater than about 1.58, the equilibrium hydrogen pressure can be approximated by:

$$P_h = \exp\left(1.76 + 10.3014x - \frac{19740.37}{T_k}\right)$$

where:

x is the ratio of hydrogen atoms to zirconium atoms, and
T_k is the fuel temperature (K).

The pressure exerted by the fission product gases is given by:

$$P_{fp} = f\frac{n}{E}\frac{RT_k}{V}E$$

where:

f is the fission product release fraction;

$\frac{n}{E}$ is the number of moles of gas evolved per unit of energy produced (mol/MW-day);
R is the gas constant (8.206×10^{-2} L-atm/mol-K);
V is the free volume occupied by the gasses (L); and
E is the total energy produced in the element (MW-day).

The fission product release fraction is given by:

$$f = 1.5\times10^{-5} + 3.6\times10^{3}\exp\left(\frac{-1.34\times10^{4}}{T_o}\right)$$

where:

T_o is the maximum fuel temperature in the element during normal operation (K).

The fission product gas production rate, $\frac{n}{E}$, varies slightly with the power density. The value 1.19×10^{-3} mol/MW-day is accurate to within a few percent over the range from a few kilowatts per element to well over 40 kW per element. The free volume occupied by the gases is assumed to be a space one-eighth inch (0.3175 cm) high at the top of the fuel so that

$$V = 0.3175\pi \cdot r_i^2$$

where:

r_i is the inside radius of the clad (1.745 cm).

For standard TRIGA fuel, the maximum burnup is about 4.5 MW-days per element, but the TRIGA-FLIP fuel is capable of burnup to about 77 MW-days per element. The LEU 30/20 fuel has been tested to 50% burnup so its capability is slightly less than FLIP fuel at 54 MW-days per element. As the fission product gas pressure is proportional to the energy released, assume that the FLIP fuel in the reactor has undergone maximum burnup.

Finally, the air trapped within the fuel element clad will exert a pressure

$$P_{air} = \frac{RT_k}{24}$$

where it is assumed that the initial specific volume of the air is 22.4 L/mol. Actually, the air forms oxides and nitrides with the zirconium, so that after relatively short operation the air is no longer present in the free volume inside the fuel element clad. The results of the stress imposed on the clad for standard and FLIP fuels are in Figure 42. The stress imposed by LEU 30/20 fuel would be slightly less than the FLIP fuel due to its lower maximum burnup capability. These results confirm the conclusion of NUREG-1282 that the LEU 30/20 fuel has a safety limit of 950 °C when the clad temperature equals the fuel temperature.

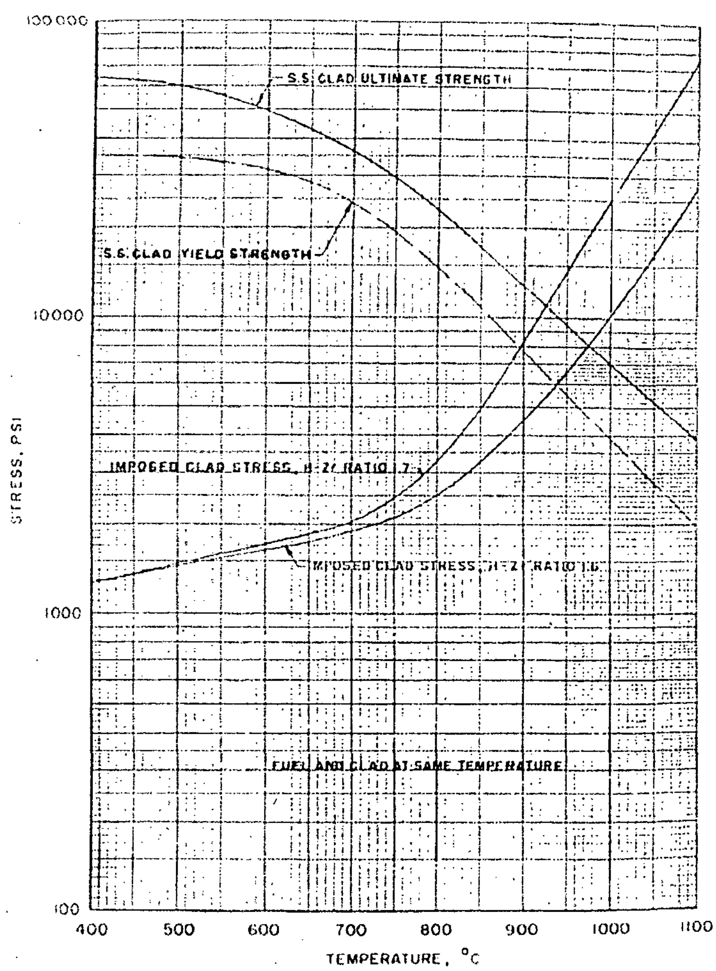

Figure 42: Strength and Applied Stress as a Function of Temperature for 1.7 and 1.6 H-Zr TRIGA Fuel

The maximum fuel cladding temperature after a loss of pool water depends on the fuel rod power density and the time delay between reactor shutdown and uncovering of the core. For the case of no delay, a value of 21 kW/element is reported to prevent fuel temperatures exceeding 900 °C (Foushee, 1972). The analysis developed a two-dimensional transient-heat transport computer code model (TAC2D) for calculating the maximum fuel temperatures after the loss of pool water for various delay times. During the loss of pool water, the low water level alarm occurs when the water level is 18 ft above the top of the core. The time between the actuation of the pool level alarm and the uncovering of the fuel for a catastrophic failure of an eight-inch beam tube is assumed to be 15 minutes (900 sec). Figure 43 presents a curve of maximum fuel cladding temperature versus fuel rod power density using a 15-minute delay. The results show that standard (8.5%) fuel can remain below 900° C at a power density of 22.3

kW/element. The results also show that a FLIP or LEU (30/20) fuel element can remain below 950° C at a power density of 23.5 kW/element. This power density is below the maximum power density of 20.8 kW/element for the WSU reactor operating at 1 MW. Additional delay time is likely since the most likely initiator of a beam tube rupture is the dropping of a large heavy object from above the reactor pool. Such activity is not anticipated for several hours after reactor shutdown.

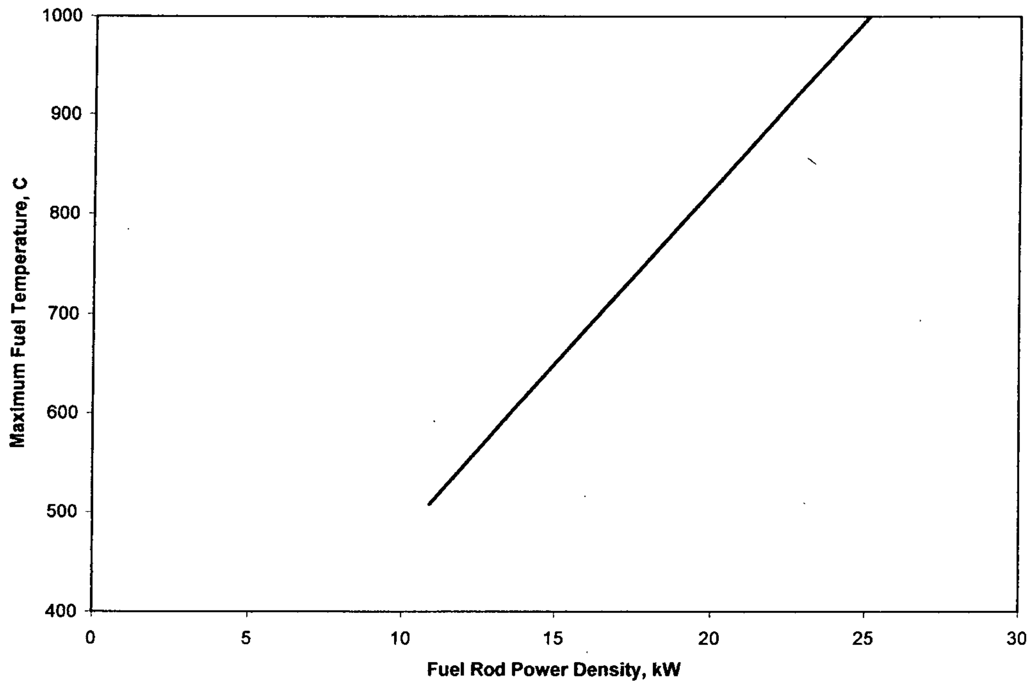

Figure 43: Maximum Fuel Rod Temperature for Loss of Coolant 15 Minutes After Shutdown

If the reactor operates for seventy MW-hours or less per week, power generation per element values approximately 20% higher are sufficient to meet the safety limits. Thus, 26.7 kW/element for standard and 28.2 kW/element for FLIP and LEU 30/20 fuel are adequate power densities. A comparison of decay heat generation versus time following loss of coolant for infinite reactor operations and 70 MW-hours per week cycle operation are in Figure 44.

Even though the probability of a loss-of-coolant accident is extremely remote, calculations have been performed to evaluate the radiological hazards. The radiation dose rates are given in Table 40 and are based on the assumption that the reactor has been operating for a very long time at a power level of 1 MW prior to the loss of pool water. The times listed in the table are after shutdown of the reactor from full power. The first location is directly on top of the reactor at the bridge level which is 23.5 feet above the top of the actual fueled portion of the core. The second location is at the pool room floor level at the freight door at the east end of the pool room. This second location is shielded from direct radiation from the core but subjected to scattered radiation from the ceiling of the pool room. The ceiling is assumed to be thick concrete yielding the maximum possible reflected radiation dose which is a conservative assumption since the actual roof structure would yield much less scattered radiation.

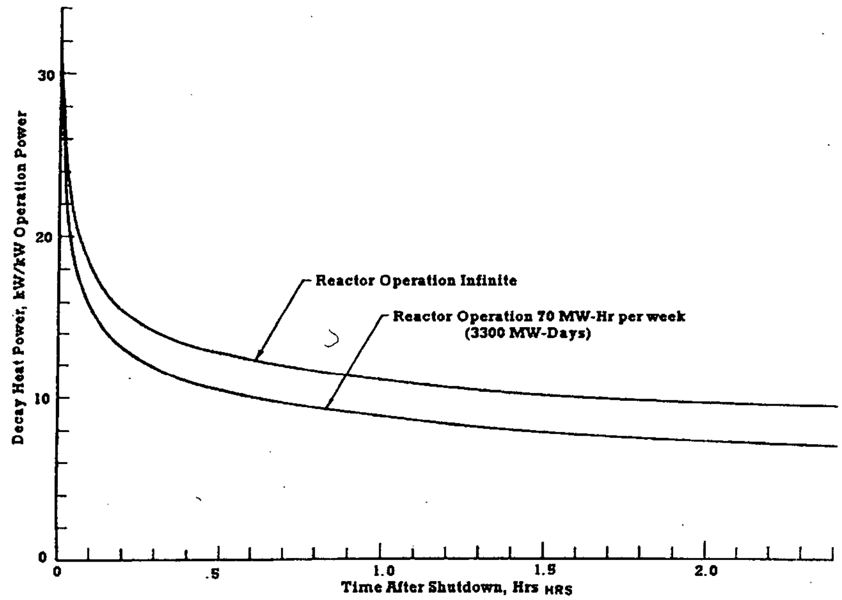

Figure 44: Decay Heat Power Generation Following Loss of Coolant for Infinite Reactor Operation and Periodic Reactor Operation

Table 40
Calculated Radiation Exposure Rates in a Loss-of-Pool-Water Accident

Time After Shutdown	Reduction Factor Due to Decay	Direct Radiation (rem/hr)	Scattered Radiation (rem/hr)
10 sec	1	7350	0.25
1 hr	2.70	2720	0.093
1 day	8.67	848	0.025
1 week	18.84	396	0.015
1 month	72.20	102	0.0035

The data given in Table 40 was calculated assuming that the bare unshielded core is a cylindrical source of 1 MeV photons with a uniform source distribution. The dimensions of the cylinder were taken equal to the active core lattice which has an equivalent radius of 29.1 cm, height of 38.1 cm, and a volume of 5.75×10^4 cm^3. The source strength as a function of time was determined from Perkins and King's data on fission product decay. No accounting was made for sources other than fission product decay gammas or for attenuation through the fuel rod end pieces, core support structure, or bridge deck plate. It is also assumed that no buildup occurs in the core. The sum total effect of these assumptions is a conservative (over estimation) of the dose rates.

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